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HEALTH PHYSICS AND SAFETY
ANNUAL REPORT FOR 1967

OAK RIDGE NATIONAL LABORATORY ∙ U.S. ATOMIC ENERGY COMMISSION

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

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HEALTH PHYSICS DIVISION

HEALTH PHYSICS AND SAFETY ANNUAL REPORT FOR 1967

K. Z. Morgan, Director

W. S. Snyder, Assistant Director

D. M. Davis, Department Head

AUGUST 1968

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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UNION CARBIDE CORPORATION
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The data for this report were contributed by: H. H. Abee, Environs Radiation Monitoring Section; R. L. Clark, Radiation and Safety Surveys Section; E. D. Gupton, Applied Radiation Dosimetry Section; A. D. Warden, Health Physics and Safety Associate Department Head; D. C. Gary, Industrial Engineering.

4.0 SUMMARY

The gaseous and liquid waste releases from the Laboratory were such that the concentration of radioactive materials in the environs was well below the maximum levels recommended by the NCRP and FRC. The average concentration of radioactive materials in the atmosphere at the X-10 site and the calculated average concentration of radioactive materials in the Clinch River at the point of entry of White Oak Creek into the River were both less than one percent of the maximum permissible as recommended in Annex 1, Table II, AEC Manual Chapter 0524.

No employee received an external or internal radiation dose which exceeded the maximum permissible levels recommended by the FRC. The highest whole body dose equivalent received by an employee was about 5.1 rem or 43 percent of the maximum permissible annual dose. No employee has a cumulative whole body dose which exceeds the recommended maximum permissible dose as based on the age proration formula 5(N-18). There were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half of a maximum permissible body burden.

During 1967, there were 16 unusual occurrences recorded, which is the lowest number recorded since the present system of reporting unusual occurrences was established in 1960. During 1966, there were 22 occurrences recorded.

The Laboratory reported four disabling injuries during 1967, which was a frequency rate of 0.50. The total number for the previous five years (1962-1966) was 51, or an average frequency rate of 1.4.

5.0 ENVIRONS MONITORING

The Health Physics Division monitors for airborne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network consists of 22 stations which are positioned in relation to ORNL operational activities (Figures 5.1 and 5.2); the perimeter air monitoring (PAM) network consists of nine stations which are located on the perimeter of the AEC controlled area (Figure 5.3); and the remote air monitoring (RAM) network consists of eight stations which are located outside the AEC controlled area at distances of from 12 to 75 miles from ORNL (Figure 5.4). The monitoring networks provide for the collection of (1) airborne radioactivity by air filtration techniques, (2) radioparticulate fallout material by impingement on gummed paper trays, and (3) rainwater for measurement of fallout occurring as rainout. The filter data are representative of radioparticulate matter which might be considered respirable; the gummed paper data are representative of radioparticulate fallout; and the rainwater data provide information on the soluble and insoluble fractions of the radioactive content of fallout material.[1]

Low-level radioactive liquid waste originating from ORNL operations are discharged, after preliminary treatment, to White Oak Creek, which is a small tributary of the Clinch River. Liquid waste releases are controlled so that the resulting average radioactive concentrations in the Clinch River are well below the maximum permissible concentrations for waste released to uncontrolled areas as recommended in Annex 1, Table II, of AEC Manual Chapter 0524.

The radioactive content of the White Oak Creek discharge is determined at White Oak Dam (Figure 5.5) which is the last control point along the stream prior to entry of White Oak Creek waters into Clinch River waters. Water samples are collected also at a number of locations along the Clinch River, beginning at a point above the entry of waste into the river via White Oak Creek and ending at Center's Ferry (near Kingston, Tennessee) about 16 miles downstream from the confluence of White Oak Creek and the Clinch River. Water samples are analyzed for gross radioactivity and for certain specified long-lived radionuclides. A weighted average maximum permissible concentration, $(MPC)_w$, for the mixture of radionuclides is calculated on the basis of the isotopic distribution in the water.

Samples of ORNL potable water are collected daily, composited and stored. At the end of each quarter these composites are analyzed radiochemically for ^{90}Sr content and are assayed for long-lived gamma emitting radionuclides by gamma spectrometry.

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[1]A detailed discussion concerning techniques used in processing air and water samples for environmental monitoring purposes is given in ORNL-2601, Radioactive Waste Management at Oak Ridge National Laboratory.

Raw milk samples are collected at 12 sampling stations located within a radius of 50 miles from ORNL. Samples are taken on a weekly basis from eight stations which are located outside the AEC controlled area within a 12-mile radius of ORNL (Figure 5.6). Samples are collected every five weeks from the four remaining stations, all of which are located outside the 12-mile radius up to distances of about 50 miles. The purpose of the milk sampling program is twofold: first, samples collected in the immediate vicinity of ORNL provide data by which one may evaluate the possible effect of waste releases originating from ORNL operations; second, samples collected remotely to the immediate vicinity of the ORNL area provide background data which are essential in establishing a proper index from which the intentional or accidental release of radioactive materials originating from Oak Ridge operations may be evaluated.

Thyroid tissues taken from cattle pastured within a radius of 100 miles of Oak Ridge are analyzed for radioiodine at the rate of six samples per week. These analyses provide information on background levels needed to identify environmental levels that might result from either continuous or sporadic releases of ^{131}I to the environment from ORNL and other Oak Ridge operations.

Aerial background surveys are made over the ORNL area and for several miles from ORNL in the general direction of low altitude prevailing winds. The frequency of flights has been established at once per quarter.

Background gamma radiation measurements are made monthly at a number of locations throughout other portions of the East Tennessee area. These measurements are taken with calibrated GM and scintillation type detectors at a distance of three feet above the surface of the ground.

River bottom sediments in the Clinch and Tennessee Rivers have been surveyed and analyzed annually since the year 1951 for the purpose of providing data relative to the dispersion of radioactive waste released from Oak Ridge operations to the Clinch River.

Fish from the Clinch River are sampled once each year and analyzed for their radioactive content. The radionuclide concentration in fish are related quantitatively to potential human intake of radioactivity through consumption of fish.

5.1 Atmospheric Monitoring

5.1.1 Air Concentrations - The average concentrations of radioactive materials in the atmosphere, as measured by filtration methods provided by the LAM, PAM, and RAM networks during 1967, were as follows:

Network	Concentration (μCi/cc)
LAM	0.22×10^{-12}
PAM	0.11×10^{-12}
RAM	0.10×10^{-12}

The LAM network value of 0.22×10^{-12} μCi/cc is about 0.02 percent of the $(MPCU)_a$[2] based on occupational exposure. Both the PAM and RAM network values represent 0.1 percent of the $(MPCU)_a$ applicable to waste released to uncontrolled areas. A tabulation of data for each station in each network is given in Table 5.1. The weekly values for each network are illustrated in Figure 5.7. The high concentration measured on the LAM network during Week No. 33 (Figure 5.7) was the result of the accidental release of approximately 1400 mCi of ^{137}Cs from the 3039 stack during this week.

The number of radioactive particles collected on the air monitoring filters is shown in Table 5.1. Data is given on both the activity range of the particles and to the total number of particles per 1000 cu. ft. of air sampled.

5.1.2 Fallout (Gummed Paper Technique) - Radioparticulate fallout as measured by the LAM network of stations increased by a factor of about 6.5 from the value measured in 1966. The values measured by the PAM and RAM networks increased by factors of 7.6 and 3.8 respectively from the 1966 values. The increase may be attributed to fallout from sources other than the Laboratory. Radioparticulate fallout showed a sharp rise in the East Tennessee area during the week ending January 8 and again during the week ending December 31. Table 5.2 gives a tabulation of data for each station within each network. The weekly average values for each network for each week are illustrated in Figure 5.8.

5.1.3 Atmospheric Radioiodine (Charcoal Collector Techniques) - Atmospheric radioiodine measured by the perimeter stations averaged 0.019×10^{-12} μCi/cc during 1967. This is only about 0.02 percent of the maximum permissible concentration applicable to waste released to uncontrolled areas. The maximum value observed at any one station for one week was 0.22×10^{-12} μCi/cc. This value was measured at PAM 33 and was associated with the release of about 400 millicuries of radioiodine from ORNL stacks[3] during a period of one week. Figure 5.9 compares the weekly discharge of radioiodine from ORNL stacks[4] with the average concentration of radioiodine measured by the perimeter stations.

The average radioiodine concentration measured by the local stations was 0.43×10^{-12} μCi/cc. This is about 0.04 percent of the maximum permissible concentration for occupational exposure. The maximum value observed on any one station for one week was 5.1×10^{-12} μCi/cc. This value was observed at LAM 4 (near the waste treatment plant). Table 5.3 gives ^{131}I data for both the Plant area (LAM's) and the perimeter area monitors.

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[2] The $(MPCU)_a$ is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. AEC Manual Chapter 0524, Appendix, Annex 1, gives exposure values applicable to various mixtures of radionuclides and establishes guide lines for deriving the $(MPCU)_a$.

[3] "Summary of Waste Discharges", week ending July 9, 1967, L. C. Lasher.

[4] "Summary of Waste Discharges", Weekly Reports, 1967, L. C. Lasher.

5.2 Water Analyses

5.2.1 Rainwater - The average concentration of radioactivity in rainwater collected from the three networks during 1967 were as follows:

Network	Concentration (μCi/ml)
LAM	0.26×10^{-7}
PAM	0.20×10^{-7}
RAM	0.19×10^{-7}

The value observed on the LAM network was essentially the same as that measured during 1966. The average values for the PAM and RAM networks are not significantly different from the average value for the LAM network. The average values for each station are shown in Table 5.4; the average values for each network for each week are given in Figure 5.10.

5.2.2 Clinch River Water - A total of 40 beta curies of radioactivity was released to the Clinch River during 1967 as compared to 48 for 1966 (Table 5.5). Yearly discharges of radionuclides to Clinch River, 1949 through 1967, are shown in Table 5.6. Radiochemical analysis of the White Oak Dam effluent indicated that about 43 percent of the radioactivity was ^{106}Ru. The percentage of ^{90}Sr in the effluent was 13 in 1967 compared to 6.0 in 1966.

The calculated average concentration of radioactive materials in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the River) was 1.2×10^{-8} μCi/ml. This represents only 0.71 percent of the weighted average $(MPC)_w$ for waste released to uncontrolled areas (Table 5.7). The average concentration of radioactive materials in the Clinch River did not exceed 3.9 percent of the $(MPC)_w$ during any week in 1967 (Figure 5.11).

The measured average concentration of radioactivity in Clinch River water at CRM 23.1 (above the entry of White Oak Creek) was 0.15 percent of the weighted average $(MPC)_w$ (Table 5.7). The concentration of ^{90}Sr in the River above the entry of White Oak Creek was about two-thirds of the contribution calculated for White Oak Creek effluent at CRM 20.8 assuming uniform mixing of the two streams.

The measured average concentration of radioactive materials in the Clinch River at CRM 4.5 (near Kingston, Tennessee) was 0.49×10^{-8} μCi/ml. This value represents 0.42 percent of the $(MPC)_w$ applicable to waste released to uncontrolled areas.

5.2.3 Potable Water - The average concentrations of ^{90}Sr in potable water at ORNL during 1967 were as follows:

Quarter Number	Concentration ^{90}Sr ($\mu Ci/ml$)
1	0.7×10^{-9}
2	0.7×10^{-9}
3	0.9×10^{-9}
4	0.9×10^{-9}
Average for Year	0.8×10^{-9}

The average value of 0.8×10^{-9} represents 0.27 percent of the $(MPC)_w$ for drinking water applicable to individuals in the general population.

Based on gamma spectrometric analyses, no long-lived gamma emitting radionuclides were detected in ORNL potable water during 1967.

5.3 Milk Analyses

The average concentration of ^{90}Sr in raw milk samples collected within a 12-mile radius of the Laboratory during 1967 was 28 pCi/l. The average concentration of ^{90}Sr in samples collected between 12 miles and 50 miles from the Laboratory was 25 pCi/l. These results would indicate that the ^{90}Sr content of milk in the Oak Ridge area is from sources other than the Laboratory. Figure 5.12 presents the weekly average concentration of ^{90}Sr in raw milk collected from the immediate environs of Oak Ridge.

The average concentration of ^{131}I in raw milk samples collected between 12 miles and 50 miles from the Laboratory during 1967 was 6.4 pCi/l. The average concentration of ^{131}I in samples collected within a 12-mile radius of the Laboratory was 17 pCi/l. Figure 5.13 presents the weekly average concentrations of ^{131}I in raw milk collected at these stations compared with the weekly discharges of ^{131}I from the ORNL stacks. It should be noted that the higher of the two yearly average concentrations is near the lower limit of FRC Range II daily intake guide for ^{131}I, if one assumes an intake of 1 liter of milk per day.

5.4 Background Measurements

Background measurements were taken at a number of locations (established in 1961) in the East Tennessee area during routine servicing visits to the remote air monitoring stations. Measurements were made at each location on a frequency of once each five weeks. The average background level during 1967 as measured at these stations was 0.011 mR/hr. Average background readings and the location of each station are presented in Figure 5.14.

Background measurements made monthly with a calibrated GM monitor at five selected locations adjacent to the ORNL area yielded an average background reading of 0.011 mR/hr during 1967. Corresponding measurements made at 53 locations on the ORNL site gave an average background of

0.071 mR/hr. The average background level measured in the Oak Ridge area in 1943 prior to the start-up of the Oak Ridge Graphite Reactor was 0.012 mR/hr. A comparison of average background values taken both on and off the X-10 site for the years 1957-67 is presented in Figure 5.15.

5.5 Annual Survey of the Clinch and Tennessee Rivers

The annual survey of the Clinch and Tennessee Rivers was carried out by the Health Physics and Safety Section during the summer of 1967. In order to determine the effect of the operation of Melton Hill Dam on the dispersal pattern of contaminated bottom sediment in the Clinch River, a more intensive survey of the Clinch River was undertaken in 1967. The survey of the Tennessee River, however, extended downstream only as far as Watts Bar Dam. The techniques and procedures used are described in ORNL-2847, Radioactivity in Silt of the Clinch and Tennessee Rivers.

The 1967 survey showed the dispersal pattern of radioactive silt in the Clinch River (Figure 5.16) to be essentially the same as in previous years. With the filling of Melton Hill Dam in May, 1962, the point of maximum gamma count rate observed apparently stabilized at CRM 11.0. Before 1962 it shifted between CRM 8.0 and CRM 11.0 being affected primarily by control operations at Norris Dam, by water level fluctuations in Watts Bar Reservoir, and by outflow conditions at Watts Bar Dam.

The average gamma count rate on bottom silt located in the Tennessee River showed a slight decrease over that of 1966* and a 40 percent decrease from the 1961 average. The count rates and the location where they were taken is given in Figure 5.17.

Radiochemical analysis data obtained from the Clinch and Tennessee River silt collected in 1966 and 1967 are given in Table 5.8 and Table 5.9.

5.6 Radionuclides in Clinch River Fish

Four species of fish were collected from the Clinch River for assay during the spring and summer of 1967. The fish were prepared for radiochemical analysis in a manner analogous to human utilization. Ten fish of each species were composited for each sample and analyzed by gamma spectrometry and radiochemistry for the critical radionuclides contributing most heavily to potential radiation dose to man.

The concentration of radionuclides found in Clinch River fish during the years 1965, 1966, and 1967 are given in Table 5.10. Smallmouth buffalo are considered to be representative of the commercial fish species and the white crappie is the most commonly caught game fish. Gizzard shad were included because of the potential utilization of this species for fish meal or cat and dog food.

The data in Table 5.10 show some of the difficulties encountered when attempting to utilize one sample a year for interpretation of results. There is sufficient variation in results to make detailed explanation

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*The 1966 river survey was not as intensive as those of 1961 and 1967 and, hence, the data are not given in Figure 5.16 and Figure 5.17.

rather tenuous. Within five to ten years, however, the results should reflect the general trend in activity releases to the river and the full value of the analyses may then be realized.

An estimate of man's intake of radionuclides from eating Clinch River fish was made by assuming an annual rate of fish consumption of 37 lbs.[5] A maximum permissible intake (MPI) was calculated by assuming a daily intake of 2.2 l of water containing the MPC_w of the radionuclide in question. The fraction of MPI attained for each radionuclide of interest was calculated from the estimated intake of contaminated fish, Table 5.11.

The highest percentage of MPI attained was 1.7 percent in 1965 from eating smallmouth buffalo. This value results from using the unusually high Ru concentration value for 1965, Table 5.10. This ^{106}Ru concentration approaches those occurring in White Oak Lake fish and is believed to be in error. If this value is neglected, all values are less than 1 percent of MPI.

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[5] "Evaluation of Radiation Dose to Man from Radionuclides Released to the Clinch River", K. E. Cowser, W. S. Snyder, C. P. McCammon, C. P. Straub, O. W. Kochtitzky, R. L. Hervin, E. G. Struxness, and R. J. Morton.

Table 5.1 Concentration of Radioactive Materials in Air—1967
(Filter Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-13} μCi/cc	No. of Particles by Activity Ranges < 10^5 d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	> 10^7 d/24 hr	Total	Particles Per 1000 ft^3
			Laboratory Area					
HP-1	S 3587	2.3	3.7	0.09	0.00	0.00	3.8	0.17
HP-2	NE 3025	3.5	4.5	0.04	0.02	0.00	4.6	0.27
HP-3	SW 1000	1.7	1.8	0.00	0.00	0.00	1.8	0.11
HP-4	W Settling Basin	1.6	2.6	0.00	0.00	0.00	2.6	0.15
HP-5	E 2506	2.3	3.3	0.09	0.00	0.00	3.4	0.20
HP-6	SW 3027	2.4	3.2	0.09	0.00	0.00	3.3	0.26
HP-7	W 7001	1.6	2.4	0.00	0.00	0.00	2.4	0.13
HP-8	Rock Quarry	1.9	3.1	0.00	0.02	0.00	3.1	0.17
HP-9	N Bethel Valley Rd.	2.1	3.2	0.00	0.00	0.02	3.2	0.18
HP-10	W 2075	2.5	5.6	0.09	0.00	0.02	5.7	0.26
HP-16	E 4500	2.5	3.9	0.08	0.00	0.00	3.9	0.33
HP-20	HFIR	1.9	2.9	0.04	0.00	0.00	2.9	0.17
Average		2.2	3.3	0.04	0.00	0.00	3.4	0.20
			Perimeter Area					
HP-31	Kerr Hollow Gate	0.98	4.5	0.00	0.00	0.00	4.5	0.08
HP-32	Midway Gate	1.6	5.2	0.02	0.02	0.02	5.3	0.09
HP-33	Gallaher Gate	0.83	3.3	0.00	0.00	0.02	3.3	0.06
HP-34	White Oak Dam	0.84	3.3	0.02	0.00	0.02	3.3	0.07
HP-35	Blair Gate	0.84	3.4	0.00	0.00	0.00	3.4	0.06
HP-36	Turnpike Gate	1.6	3.6	0.68	0.00	0.00	4.3	0.09
HP-37	Hickory Creek Bend	0.67	3.3	0.00	0.00	0.00	3.3	0.06
HP-38	E EGCR	1.3	4.3	0.04	0.00	0.02	4.4	0.11
HP-39	Townsite	1.3	4.6	0.00	0.00	0.02	4.6	0.13
Average		1.1	3.9	0.08	0.00	0.01	4.0	0.08
			Remote Area					
HP-51	Norris Dam	1.2	5.4	0.02	0.00	0.00	5.4	0.10
HP-52	Loudoun Dam	0.95	5.3	0.02	0.00	0.00	5.3	0.09
HP-53	Douglas Dam	0.97	2.5	0.02	0.00	0.02	2.5	0.05
HP-54	Cherokee Dam	0.88	4.5	0.02	0.00	0.00	4.5	0.08
HP-55	Watts Bar Dam	1.2	4.3	0.00	0.00	0.00	4.3	0.08
HP-56	Great Falls Dam	0.90	3.0	0.00	0.00	0.00	3.0	0.06
HP-57	Dale Hollow Dam	0.96	4.6	0.00	0.00	0.00	4.6	0.09
HP-58	Knoxville	1.1	4.6	0.00	0.00	0.00	4.6	0.10
Average		1.0	4.3	0.01	0.00	0.00	4.3	0.08

Table 5.2 Radioparticulate Fallout—1967
(Gummed Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-4} $\mu Ci/ft^2$	No. of Particles by Activity Ranges $< 10^5$ d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	$> 10^7$ d/24 hr	Total Particles Per Sq. Ft.
			Laboratory Area				
HP-1	S 3587	1.2	10.5	0.15	0.00	0.00	10.6
HP-2	NE 3025	2.5	12.2	0.11	0.02	0.02	12.4
HP-3	SW 1000	1.3	11.7	0.17	0.00	0.02	11.9
HP-4	W Settling Basin	1.7	11.5	0.28	0.00	0.00	11.8
HP-5	E 2506	1.8	15.0	0.11	0.00	0.00	15.2
HP-6	SW 3027	1.8	11.4	0.25	0.00	0.00	11.7
HP-7	W 7001	1.3	15.3	0.00	0.00	0.00	15.3
HP-8	Rock Quarry	1.5	16.3	0.06	0.02	0.00	16.4
HP-9	N Bethel Valley Rd.	1.2	10.5	0.04	0.00	0.02	10.5
HP-10	W 2075	2.1	13.8	0.13	0.02	0.00	14.0
*HP-16	E 4500	1.4	12.5	0.15	0.00	0.00	12.6
*HP-20	HFIR	1.3	11.3	0.09	0.00	0.00	11.4
Average		1.6	12.7	0.43	0.00	0.00	12.8
			Perimeter Area				
HP-31	Kerr Hollow Gate	1.5	15.8	0.00	0.00	0.00	15.8
HP-32	Midway Gate	1.6	13.9	0.00	0.00	0.00	13.9
HP-33	Gallaher Gate	1.2	12.7	0.00	0.00	0.00	12.7
HP-34	White Oak Dam	0.25	8.0	0.00	0.00	0.04	8.0
HP-35	Blair Gate	0.99	8.4	0.00	0.00	0.00	8.4
HP-36	Turnpike Gate	1.6	14.4	0.04	0.00	0.00	14.4
HP-37	Hickory Creek Bend	1.1	9.8	0.04	0.00	0.00	9.8
HP-38	E EGCR	1.3	12.5	0.00	0.00	0.00	12.5
HP-39	Townsite	1.8	15.6	0.02	0.00	0.00	15.6
Average		1.3	12.3	0.01	0.00	0.00	12.3
			Remote Area				
HP-51	Norris Dam	0.96	8.4	0.02	0.00	0.00	8.4
HP-52	Loudoun Dam	0.91	7.1	0.04	0.00	0.00	7.1
HP-53	Douglas Dam	0.89	8.2	0.02	0.00	0.00	8.2
HP-54	Cherokee Dam	1.0	10.8	0.00	0.00	0.00	10.8
HP-55	Watts Bar Dam	0.42	2.9	0.04	0.00	0.00	2.4
HP-56	Great Falls Dam	0.33	2.3	0.00	0.00	0.00	2.3
HP-57	Dale Hollow Dam	0.35	2.0	0.00	0.00	0.00	2.0
HP-58	Knoxville	1.1	9.0	0.06	0.00	0.00	9.1
Average		0.75	6.3	0.02	0.00	0.00	6.3

*Installed July, 1966.

Table 5.3 Concentration of ^{131}I in Air—1967

Location	Units of 10^{-12} μCi/cc		
	Maximum	Minimum[a]	Average
ORNL Plant Area	5.1	< 0.020	0.432
Perimeter Area	0.22	< 0.010	0.019

[a]Minimum detectable amount of ^{131}I is 20 d/m. At the average sampling rate this corresponds to approximately 0.010 x 10^{-12} μCi/cc on the perimeter monitors and approximately 0.020 x 10^{-12} μCi/cc on the Plant monitors. In averaging, one-half of this value, 10 d/m is used for all samples showing a total amount of ^{131}I less than 20 d/m.

Table 5.4 Concentration of Radioactive Materials in Rainwater—1967 (Weekly Average by Stations)

Station Number	Location	Activity in Collected Rainwater, μCi/ml
	Laboratory Area	
HP-7	West 7001	0.26×10^{-7} μCi/ml
	Perimeter Area	
HP-31	Kerr Hollow Gate	0.18×10^{-7} μCi/ml
HP-32	Midway Gate	0.31
HP-33	Gallaher Gate	0.17
HP-34	White Oak Dam	0.17
HP-35	Blair Gate	0.17
HP-36	Turnpike Gate	0.13
HP-37	Hickory Creek Bend	0.14
HP-38	E EGCR	0.27
HP-39	Townsite	0.25
Average		0.20×10^{-7} μCi/ml
	Remote Area	
HP-51	Norris Dam	0.23×10^{-7} μCi/ml
HP-52	Loudoun Dam	0.20
HP-53	Douglas Dam	0.24
HP-54	Cherokee Dam	0.24
HP-55	Watts Bar Dam	0.13
HP-56	Great Falls Dam	0.12
HP-57	Dale Hollow Dam	0.17
HP-58	Knoxville	0.22
Average		0.19×10^{-7} μCi/ml

Table 5.5 Liquid Waste Discharged from White Oak Creek—1967

	Curies	
	Total for Year	Weekly Average
Beta Activity	40	0.75
Transuranic Alpha Emitters	1.0	0.018

Table 5.6 Yearly Discharges of Radionuclides to Clinch River (Curies)

Year	Gross Beta	^{137}Cs	^{106}Ru	^{90}Sr	TRE*(-Ce)	^{144}Ce	^{95}Zr	^{95}Nb	^{131}I	^{60}Co
1949	718	77	110	150	77	18	180	22	77	
1950	191	19	23	38	30		15	42	19	
1951	101	20	18	29	11		4.5	2.2	18	
1952	214	9.9	15	72	26	23	19	18	20	
1953	304	6.4	26	130	110	6.7	7.6	3.6	2.1	
1954	384	22	11	140	160	24	14	9.2	3.5	
1955	437	63	31	93	150	85	5.2	5.7	7.0	6.6
1956	582	170	29	100	140	59	12	15	3.5	46
1957	397	89	60	83	110	13	23	7.1	1.2	4.8
1958	544	55	42	150	240	30	6.0	6.0	8.2	8.7
1959	937	76	520	60	94	48	27	30	0.5	77
1960	2190	31	1900	28	48	27	38	45	5.3	72
1961	2230	15	2000	22	24	4.2	20	70	3.7	31
1962	1440	5.6	1400	9.4	11	1.2	2.2	7.7	0.36	14
1963	470	3.5	430	7.8	9.4	1.5	0.34	0.71	0.44	14
1964	234	6.0	191	6.6	13	0.3	0.16	0.07	0.29	15
1965	95	2.1	69	3.4	5.9	0.1	0.33	0.33	0.20	12
1966	48	1.6	29	3.0	4.9	0.1	0.67	0.67	0.24	7
1967	40	2.7	17	5.1	8.5	0.2	0.49	0.49	0.91	3

*Tri-Valent Rare Earths.

Table 5.7 Radioactivity in Clinch River—1967

Location	Concentration of Radionuclides of Primary Concern in Units of 10^{-8} µCi/ml						Average Concentration of Total Radioactivity	$(MPC)_w$ [a]	% of $(MPC)_w$
	^{90}Sr	^{144}Ce	^{137}Cs	$^{103-106}Ru$	^{60}Co	^{95}Zr-^{95}Nb	10^{-8} µCi/ml	10^{-6} µCi/ml	
CRM 23.1[b]	0.04	0.01	<0.01	0.03	*	< 0.01	0.09	0.61	0.15
CRM 20.8[c]	0.06	<0.01	0.02	0.20	0.03	< 0.01	1.2	1.7	0.71
CRM 4.5[b]	0.12	0.02	0.15	0.06	0.12	0.01	0.49	1.1	0.42

[a]Weighted average $(MPC)_w$ calculated for the mixture, using $(MPC)_w$ values for specific radionuclides specified by AEC Manual, Chapter 0524, Appendix, Annex 1, Table II.

[b]Measured values.

[c]Values given for this location are calculated values based on the levels of waste released and the dilution afforded by the river; they do not include amounts of radioactive material (e.g., fallout) that may enter the river upstream from CRM 20.8.

*None detected.

Table 5.8 Radionuclides in Clinch River Silt—1961 and 1967
(Units of 10^{-6} μCi/g of Dried Silt)

Location	^{137}Cs		^{144}Ce		^{90}Sr		^{60}Co		$^{103-106}Ru$		$^{95}Zr + {}^{95}Nb$	
	1961	1967	1961	1967	1961	1967	1961	1967	1961	1967	1961	1967
CRM 21.5	1.3	0.46	0.44	0.18	0.26	0.36	0.32	0.24	2.7	0.29	0.50	0.02
20.7	95	660	3.6	12	0.86	2.4	13	68	85	0.83	1.5	1.9
19.1	41	2.6	2.7	0.30	1.0	0.63	5.9	0.44	95	0.37	1.4	0.03
17.8	44	78	2.8	1.2	0.90	1.4	5.9	2.8	58	0.69	1.4	0.63
16.3	71	58	5.6	1.2	2.0	1.1	11	7.1	159	0.82	1.7	0.01
15.2	64	105	4.4	1.1	0.77	1.9	10	5.9	148	2.0	1.8	0.89
14.0	127	55	8.2	1.6	1.1	0.63	14	5.7	153	2.2	1.8	0.02
12.5	78	61	3.7	1.1	1.3	1.2	14	6.7	216	3.0	0.77	0.01
11.0	98	53	9.4	1.3	1.0	1.0	14	5.5	144	2.1	4.6	0.10
9.4	87	50	6.7	2.1	0.81	0.86	12	7.2	127	2.9	3.0	0.05
8.0	81	73	9.4	1.4	1.4	0.68	11	11	152	3.7	2.3	0.04
6.9	83	66	6.4	1.4	1.3	0.54	12	8.2	105	2.5	2.3	0.05
5.8	115	90	9.9	1.8	1.0	0.54	14	12	157	5.0	2.7	0.06
4.7	112	72	9.9	1.9	1.3	0.72	15	11	148	4.4	2.7	0.09
2.6	82	41	7.0	1.4	0.90	0.63	11	6.8	103	3.5	1.1	0.05
1.1	100	61	8.6	1.7	0.41	0.50	12	8.7	141	3.5	1.8	0.07
Average	80	95	6.2	2.0	1.0	0.94	10	10	125	2.4	2.0	0.25
Melton Hill Background Data												
CRM 42.8	*	1.3	*	0.34	*	0.45	*	0.44	*	0.53	*	0.30
34.7	*	3.1	*	0.68	*	0.36	*	0.39	*	1.4	*	0.25
31.1	*	2.9	*	0.79	*	0.36	*	0.34	*	1.2	*	0.34
24.9	*	2.9	*	0.60	*	0.27	*	0.46	*	1.4	*	0.38
Average		2.6		0.60		0.36		0.41		1.1		0.32

*No samples taken at these locations in 1961. Melton Hill Reservoir filled in May, 1962.

Table 5.9 Radionuclides in Tennessee River Silt—1961 and 1967
(Units of 10^{-6} μCi/g of Dried Silt)

Location	^{137}Cs		^{144}Ce		^{90}Sr		^{60}Co		$^{103-106}Ru$		$^{95}Zr + ^{95}Nb$	
	1961	1967	1961	1967	1961	1967	1961	1967	1961	1967	1961	1967
TRM 570.8	1.1	1.8	0.72	0.55	0.14	0.59	0.36	0.21	2.7	1.6	0.72	0.03
562.7	34	19	2.9	0.86	0.49	0.32	4.5	2.5	34	2.1	1.2	0.16
552.7	42	18	2.9	0.82	0.59	0.36	5.3	2.2	39	1.8	1.4	0.32
543.8	15	11	1.1	1.0	0.27	0.86	2.5	1.6	22	2.4	1.1	0.20
532.0	25	11	2.3	0.69	0.63	0.54	3.9	1.7	31	1.4	0.86	0.36
Average	23	12	2.0	0.78	0.42	0.53	3.3	1.6	26	1.9	1.1	0.21
Fort Loudoun Background Data												
TRM 615.8	1.2	1.9	0.77	0.56	0.14	0.32	0.36	0.06	1.9	0.93	0.72	0.29
604.4	0.99	1.8	0.99	0.48	0.18	0.63	0.32	0.03	2.3	0.78	0.68	0.17
Average	1.1	1.9	0.88	0.52	0.16	0.48	0.34	0.05	2.1	0.86	0.70	0.23

Table 5.10 Radionuclide Content of Clinch River Fish

		pCi/kg Fresh Weight			
Species	Year	^{90}Sr	^{106}Ru	^{137}Cs	^{65}Zn
White Crappie	1965	14	284	199	*
	1966	9.4	381	87	*
	1967	27	*	387	*
Smallmouth Buffalo	1965	32	6467	194	*
	1966	94	185	1303	232
	1967	27	122	402	*
Carp	1966	62	*	397	*
Gizzard Shad	1966	2028	513	1453	*
	1967	118	*	399	40

*None detected.

Table 5.11 Estimated Percentage of MPI That Man May Attain by Eating Clinch River Fish

Smallmouth Buffalo

Year	^{90}Sr	^{106}Ru	^{137}Cs	Total (%)
1965	0.24	1.4	0.020	1.66
1966	0.66	0.039	0.137	0.836
1967	0.19	0.026	0.044	0.260

White Crappie

Year	^{90}Sr	^{106}Ru	^{137}Cs	Total (%)
1965	0.099	0.060	0.021	0.180
1966	0.066	0.082	0.009	0.157
1967	0.19	*	0.041	0.231

*None detected.

ORNL-DWG. 66-2218

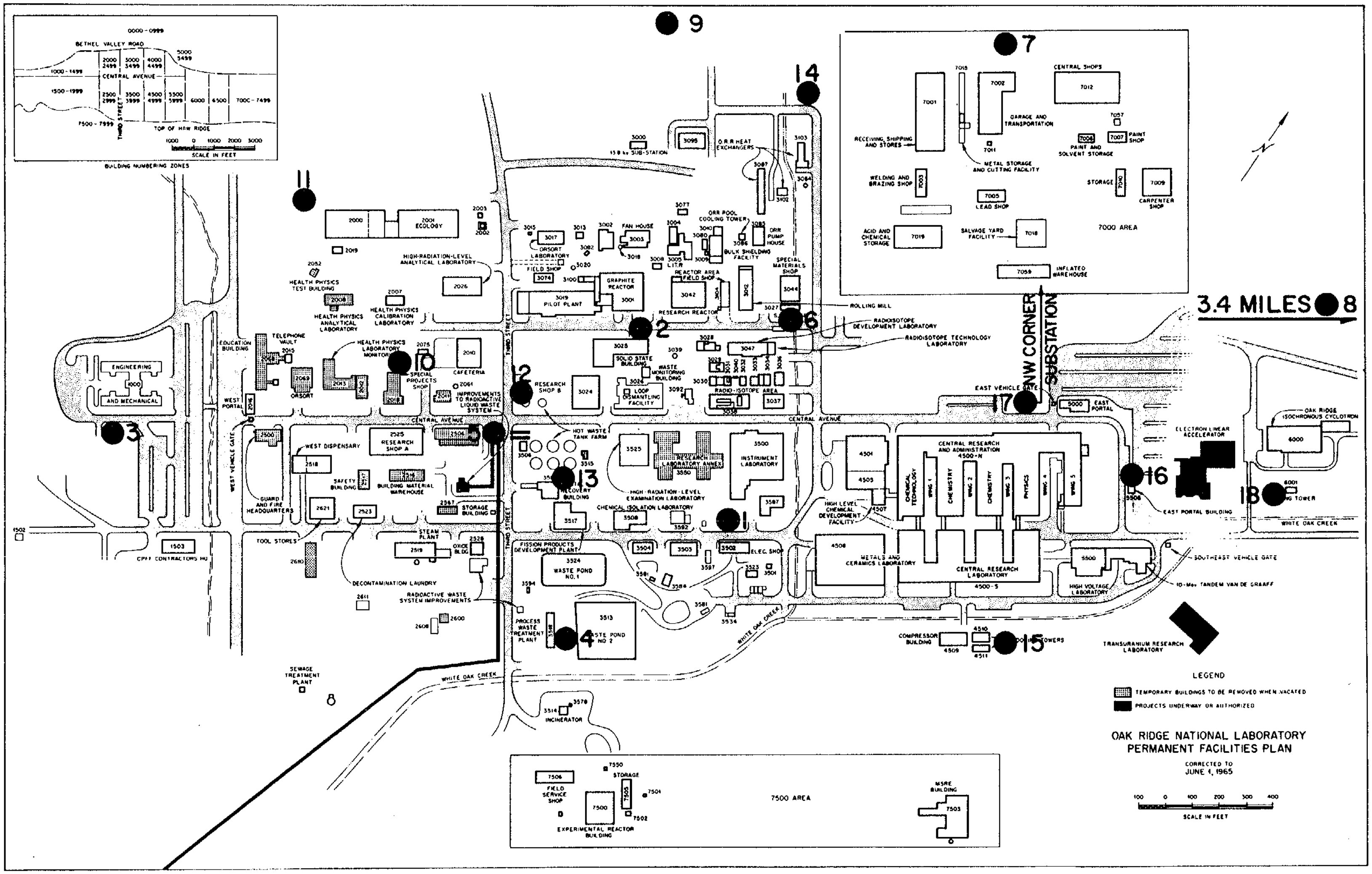

Fig. 5.1 Map of X-10 Area Showing the Approximate Location of 18 to 22 of the Local Monitoring Stations Constituting the LAM Network.

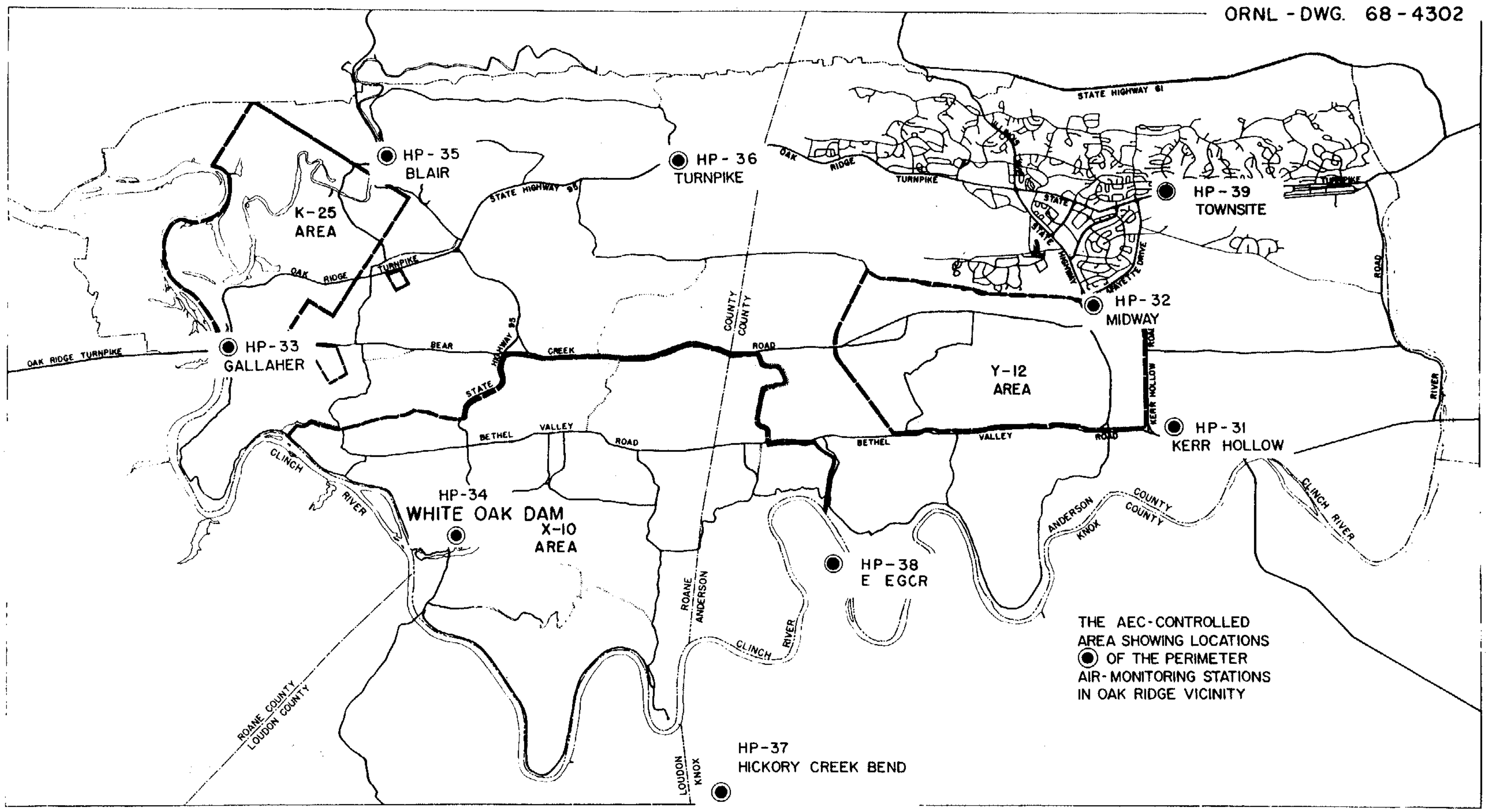

Fig. 5.3 Map of the AEC Controlled Area and Vicinity Showing the Approximate Location of the Perimeter Air Monitoring Stations Constituting the PAM Network.

ORNL-DWG. 66-1718

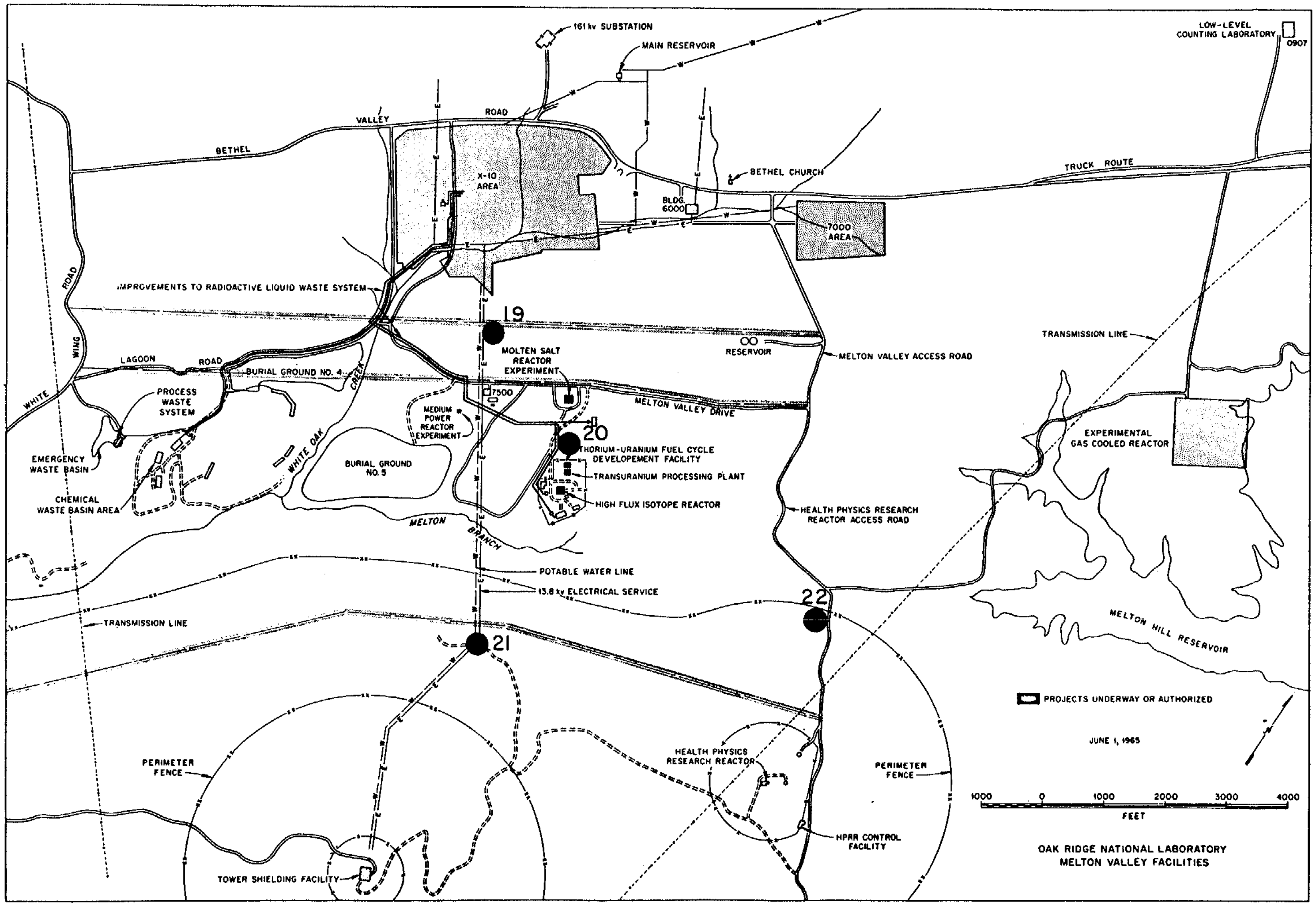

Fig. 5.2 Map of Laboratory Area Showing the Approximate Location of 4 of the 22 Local Monitoring Stations Constituting the LAM Network

ORNL-DWG 66-1719

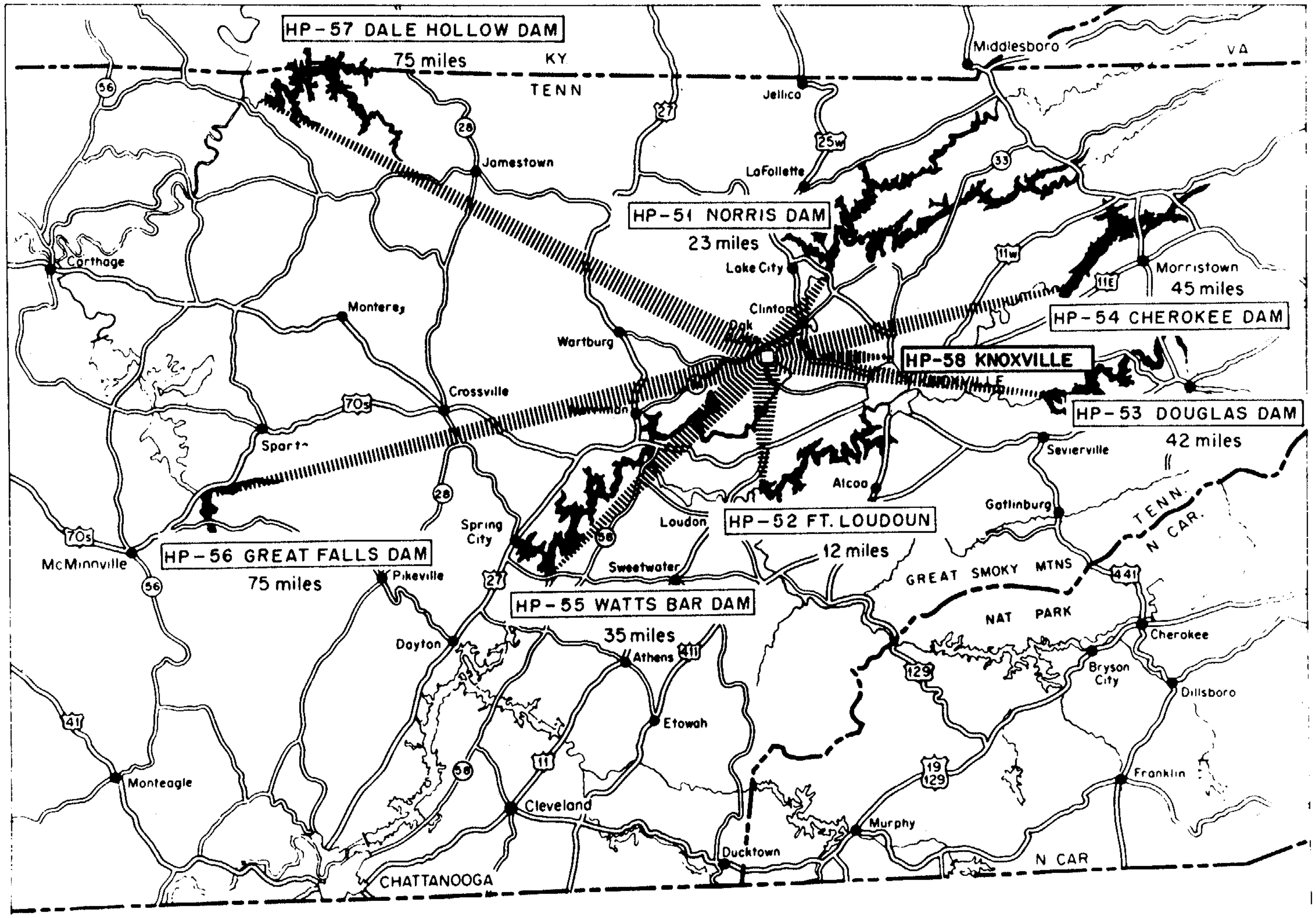

Fig. 5.4 Map of a Section of the East Tennessee Area Showing TVA and U. S. Corps of Engineers Dam Sites at which are Located the Remote Air Monitoring Stations Constituting the RAM Network.

ORNL-DWG. 66-2216R

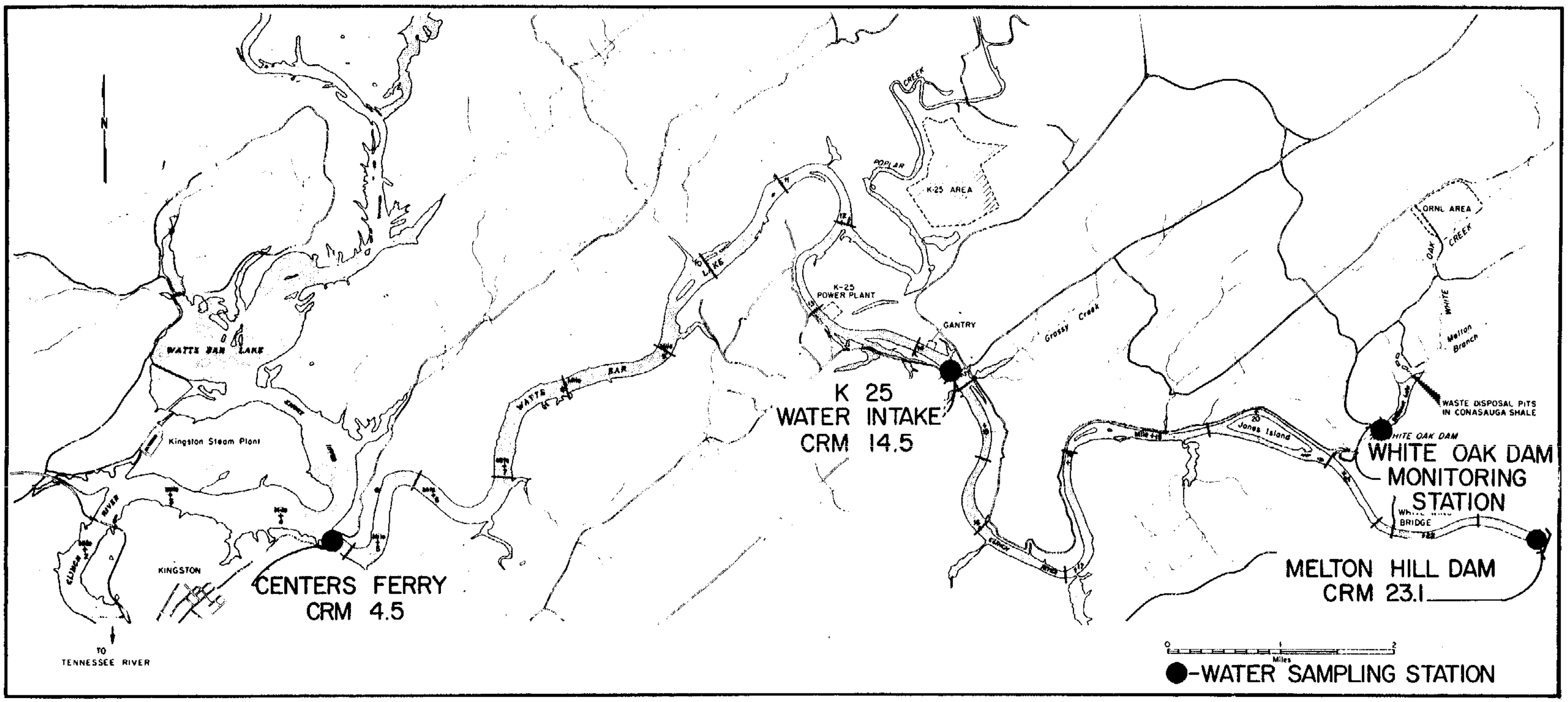

Fig. 5.5 Map Showing Water Sampling Locations in the East Tennessee Area.

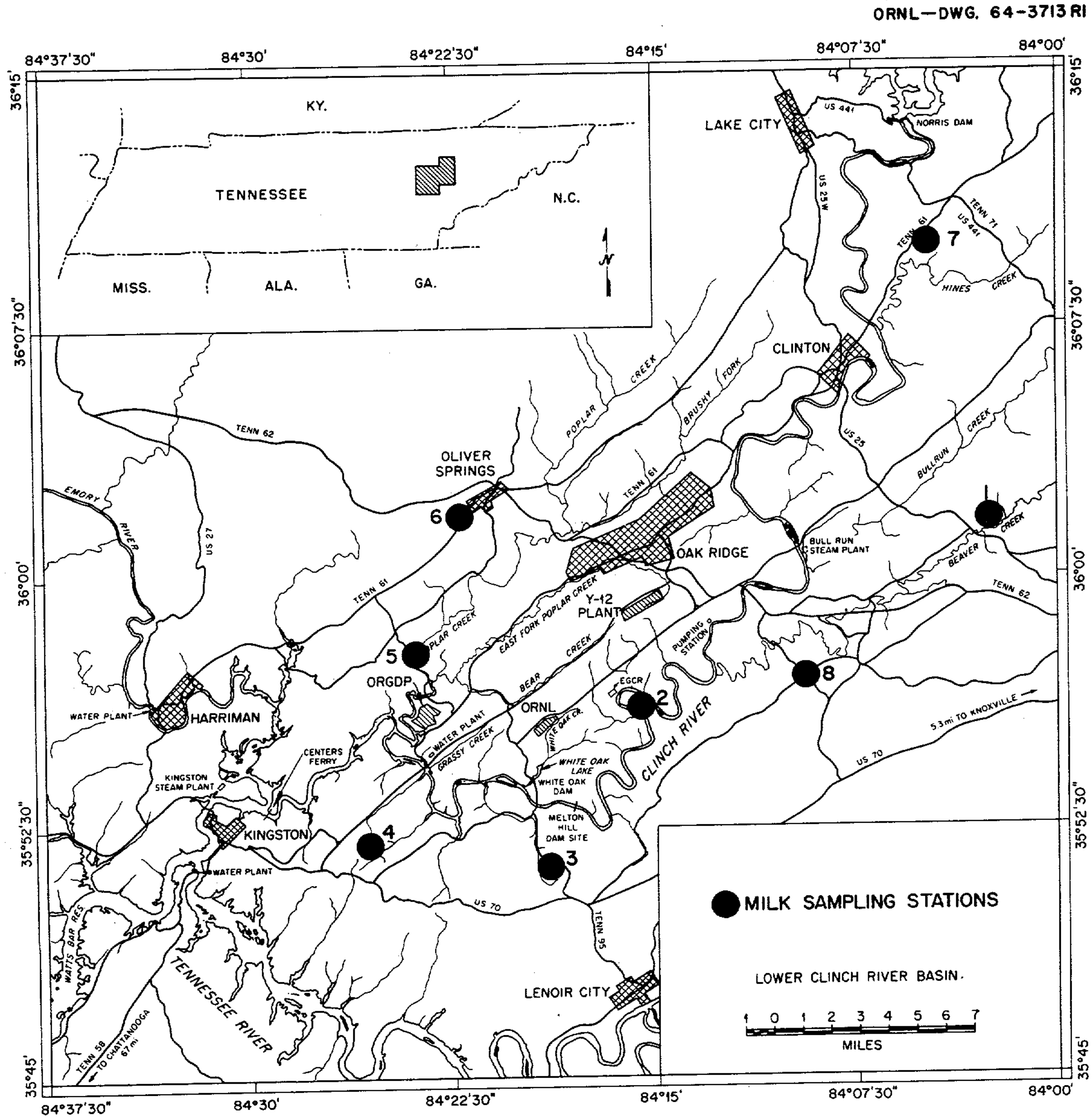

Fig. 5.6 Map Showing Milk Sampling Stations in the East Tennessee Area.

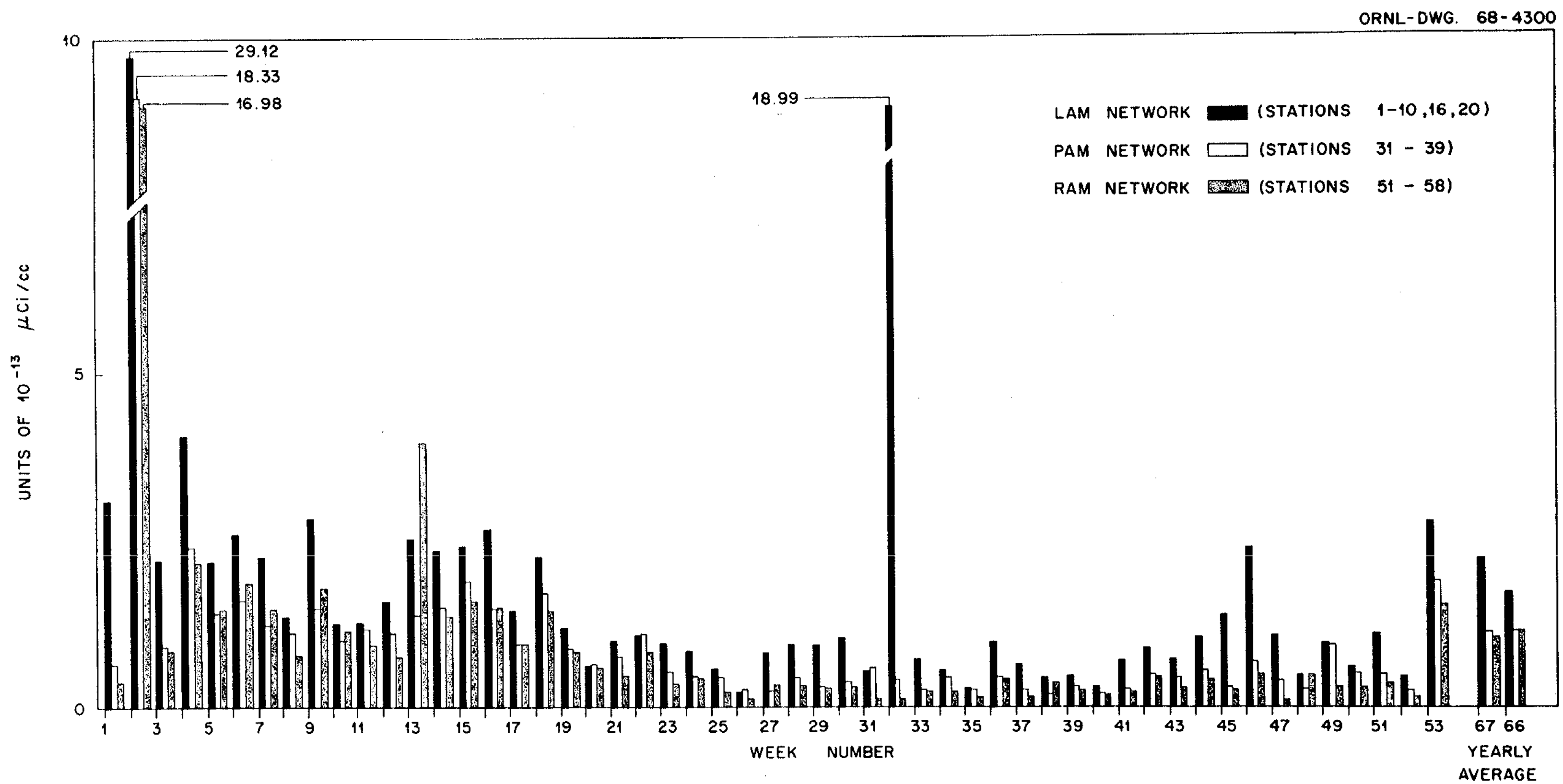

Fig. 5.7 Concentration of Radioactive Materials in Air as Determined from Filter Paper Data - 1967.

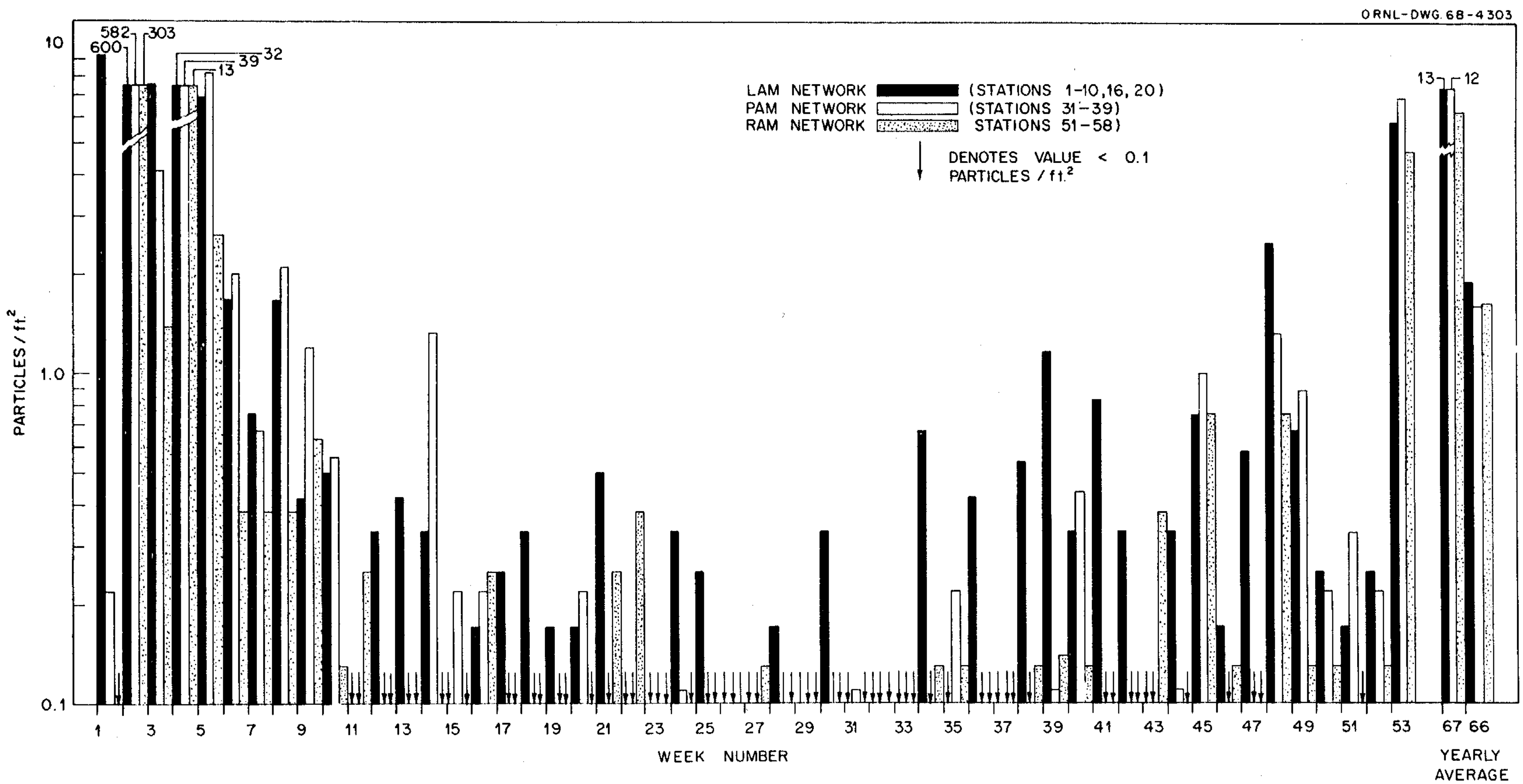

Fig. 5.8 Radioparticulate Fallout Measurements as Determined by Autoradiographic Techniques Using Gummed Paper Collectors - 1967.

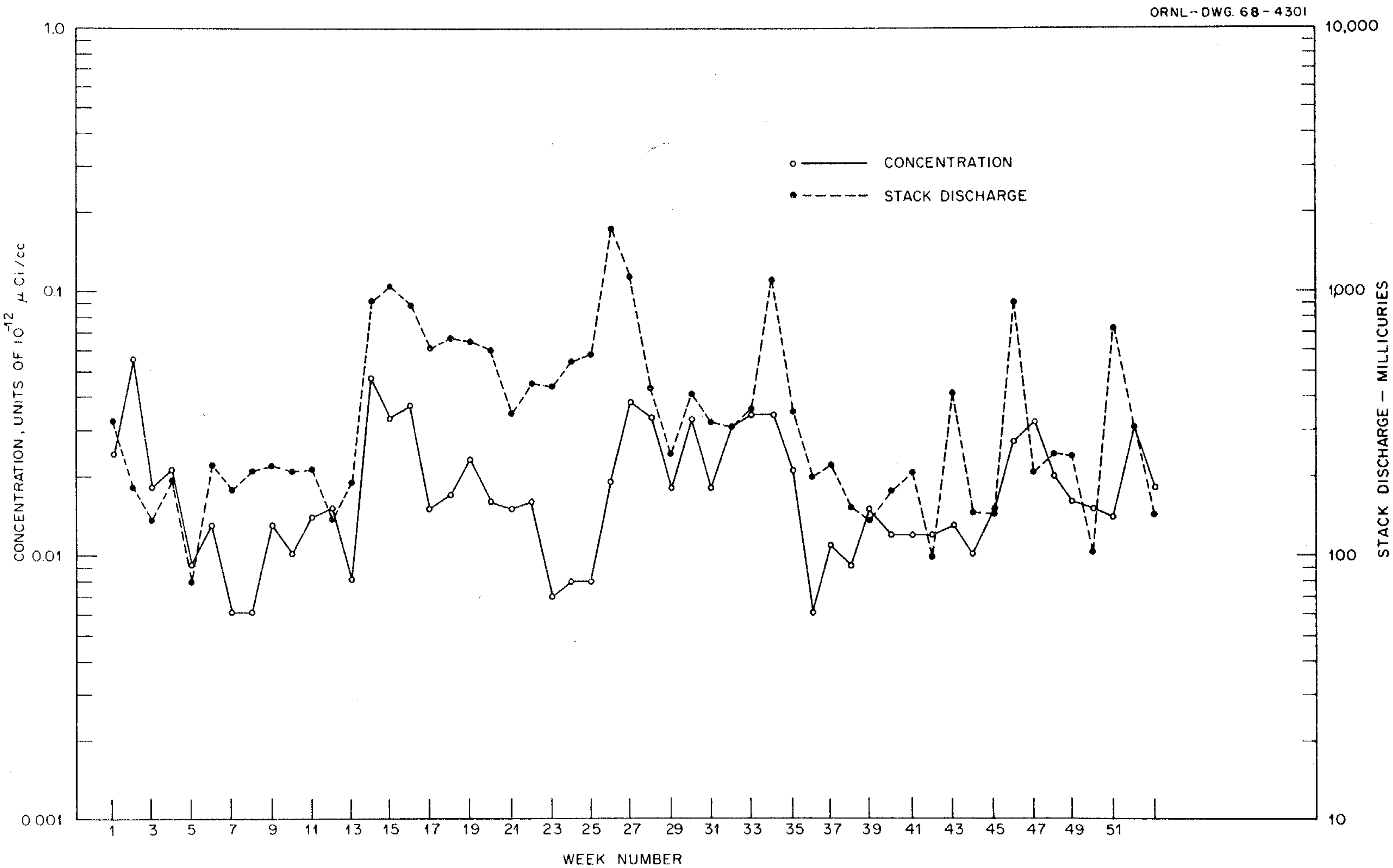

Fig. 5.9 Weekly Average Concentration of ^{131}I in Air at the Perimeter of the Controlled Area Compared with ^{131}I Discharges from ORNL Stacks - 1967.

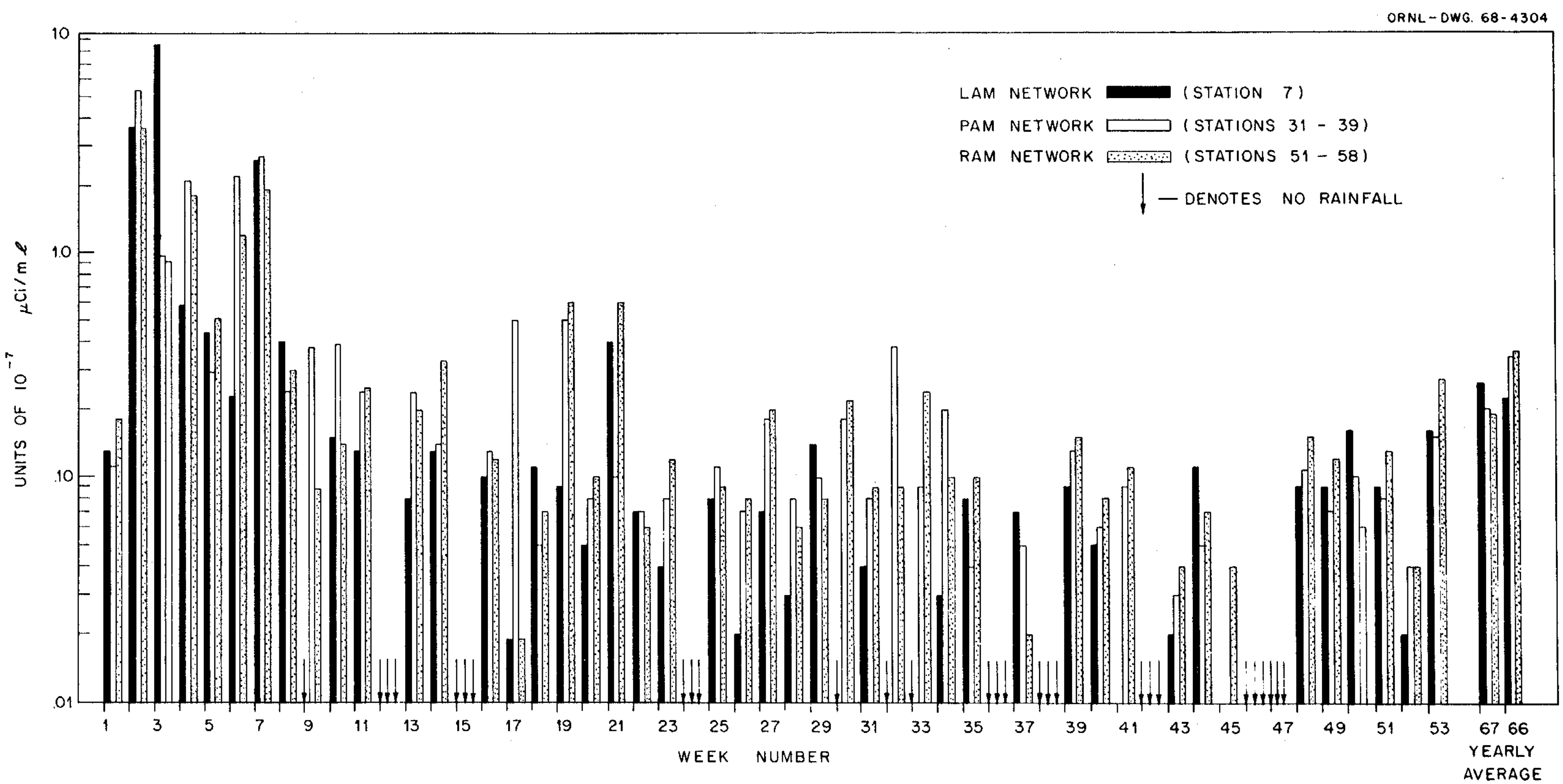

Fig. 5.10 Concentration of Radioactive Materials in Rainwater - 1967.

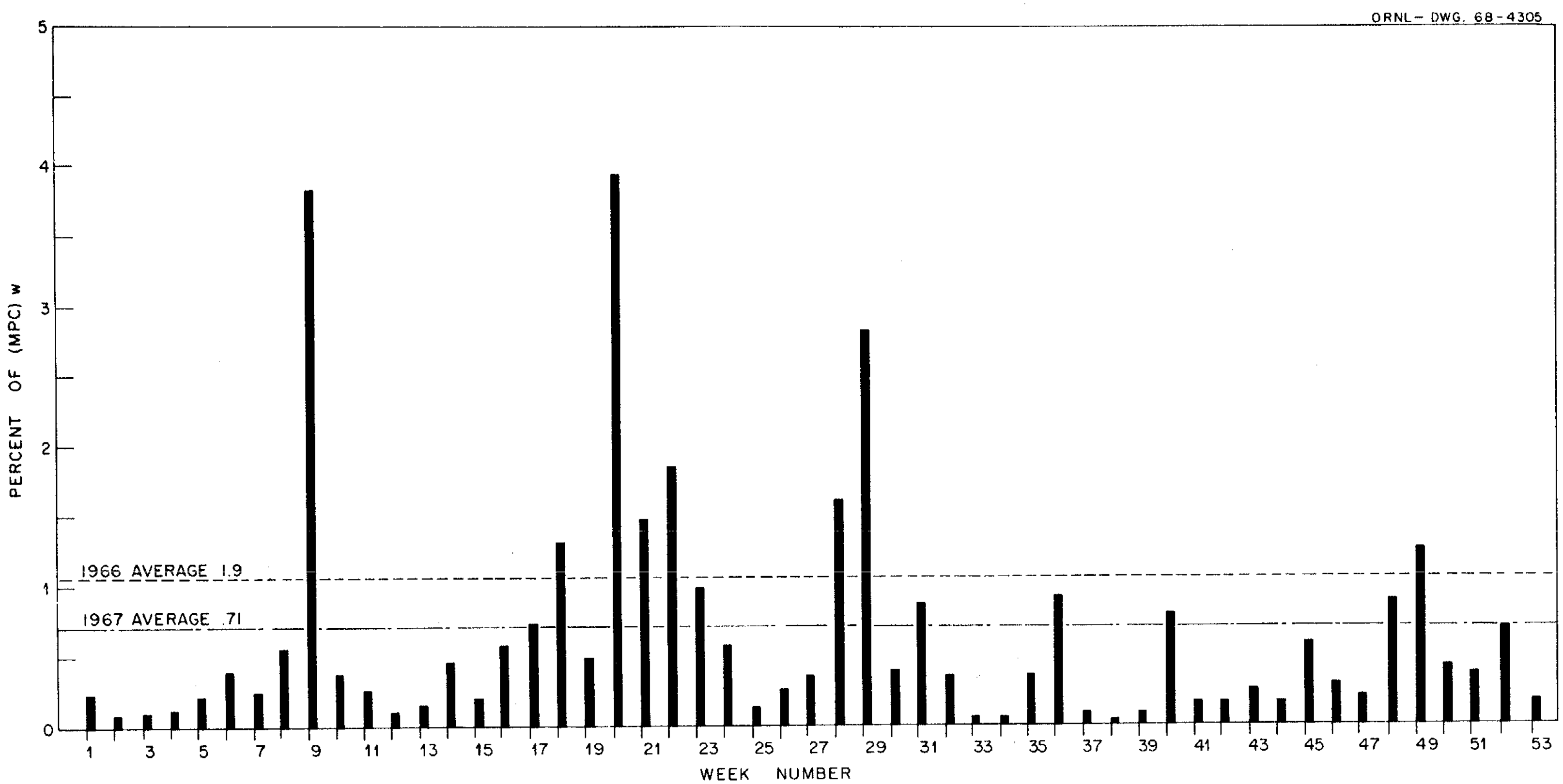

Fig. 5.11 Estimated Percent $(MPC)_w$ of Radioactivity in Clinch River Water Below the Mouth of White Oak Creek - 1967.

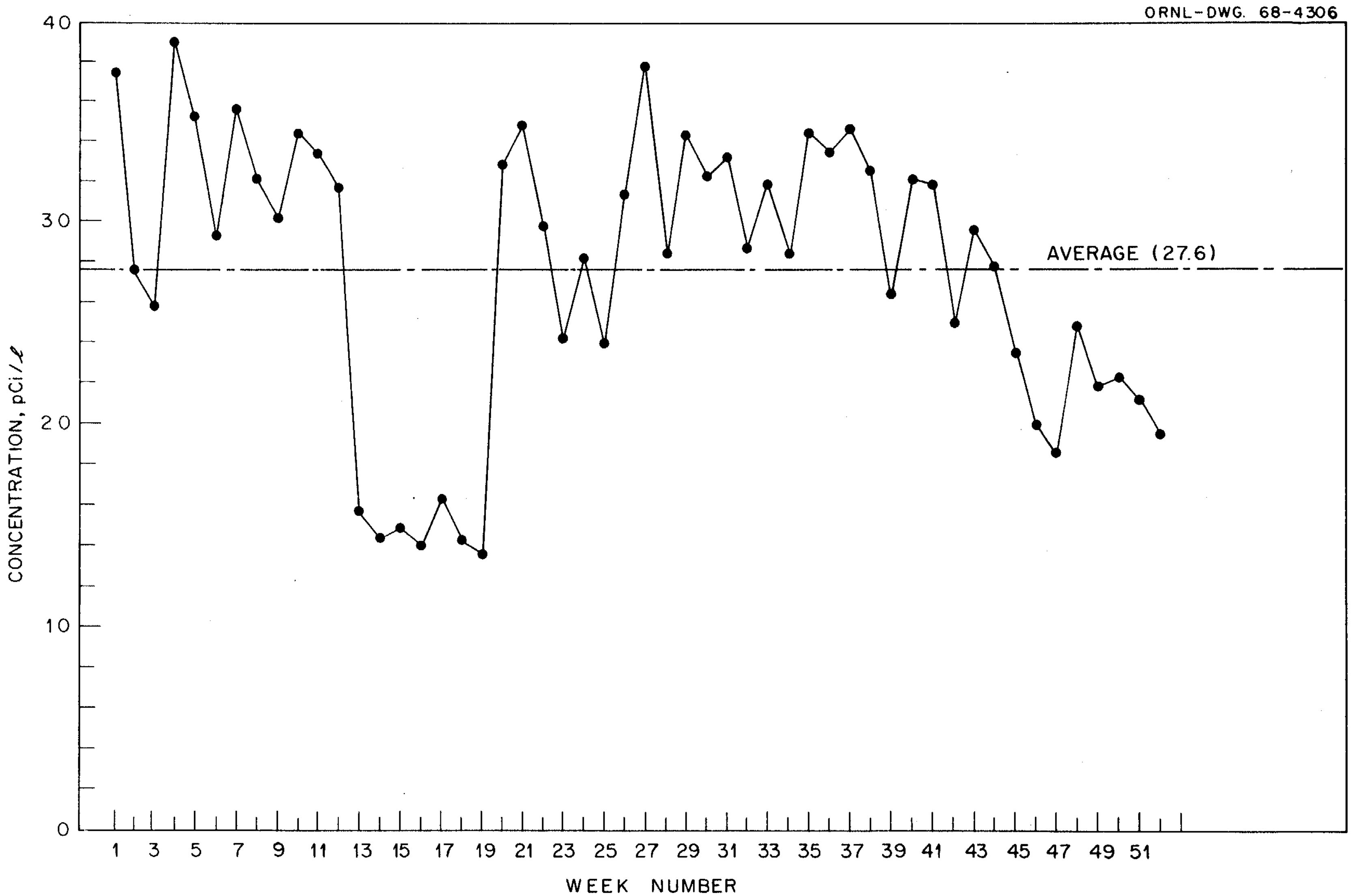

Fig. 5.12 Weekly Average Concentration of ^{90}Sr in Raw Milk in the Immediate Environs of Oak Ridge - 1967.

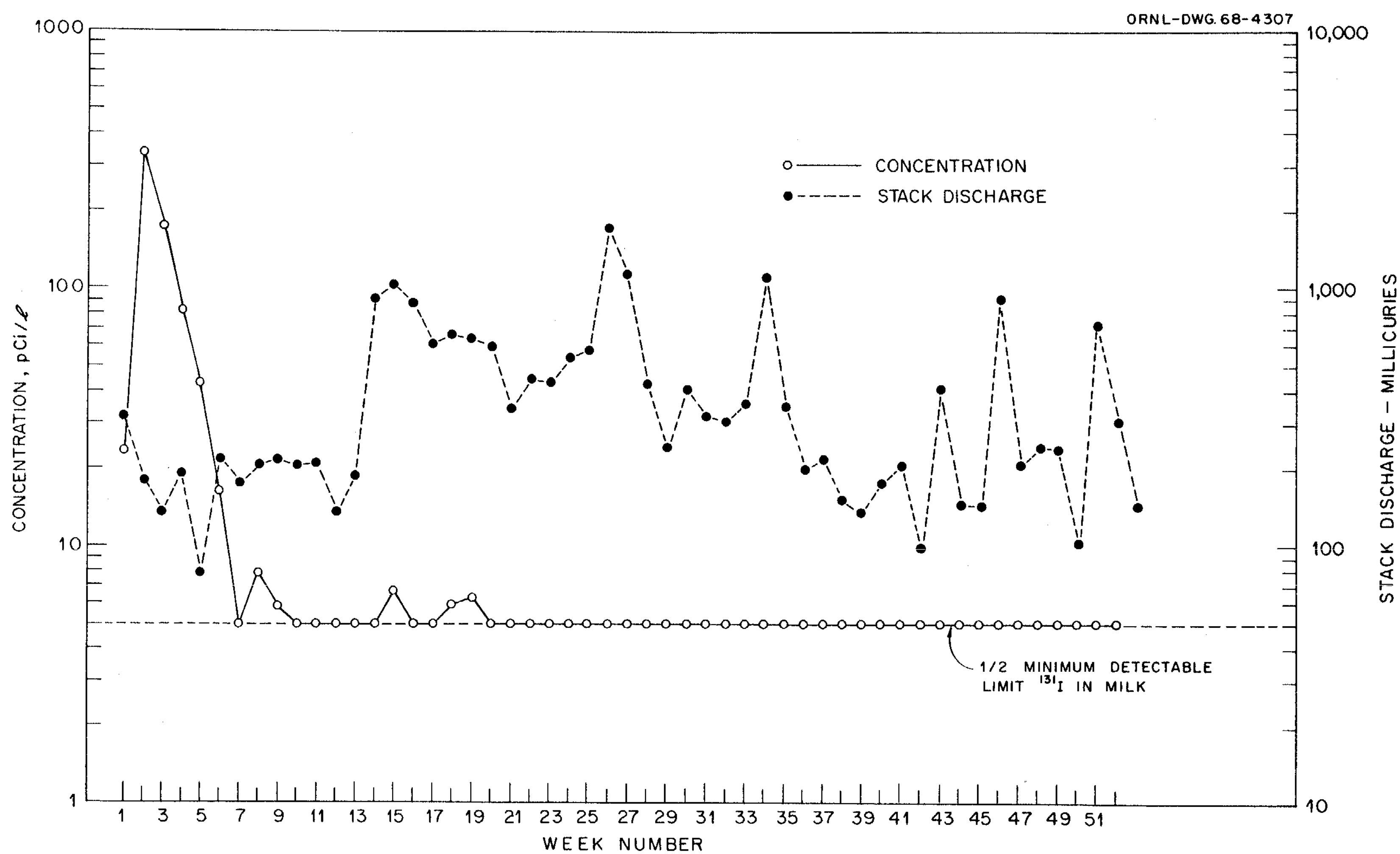

Fig. 5.13 Weekly Average Concentration of ^{131}I in Raw Milk in the Immediate Environs of Oak Ridge Compared with ^{131}I Discharges from ORNL Stacks - 1967.

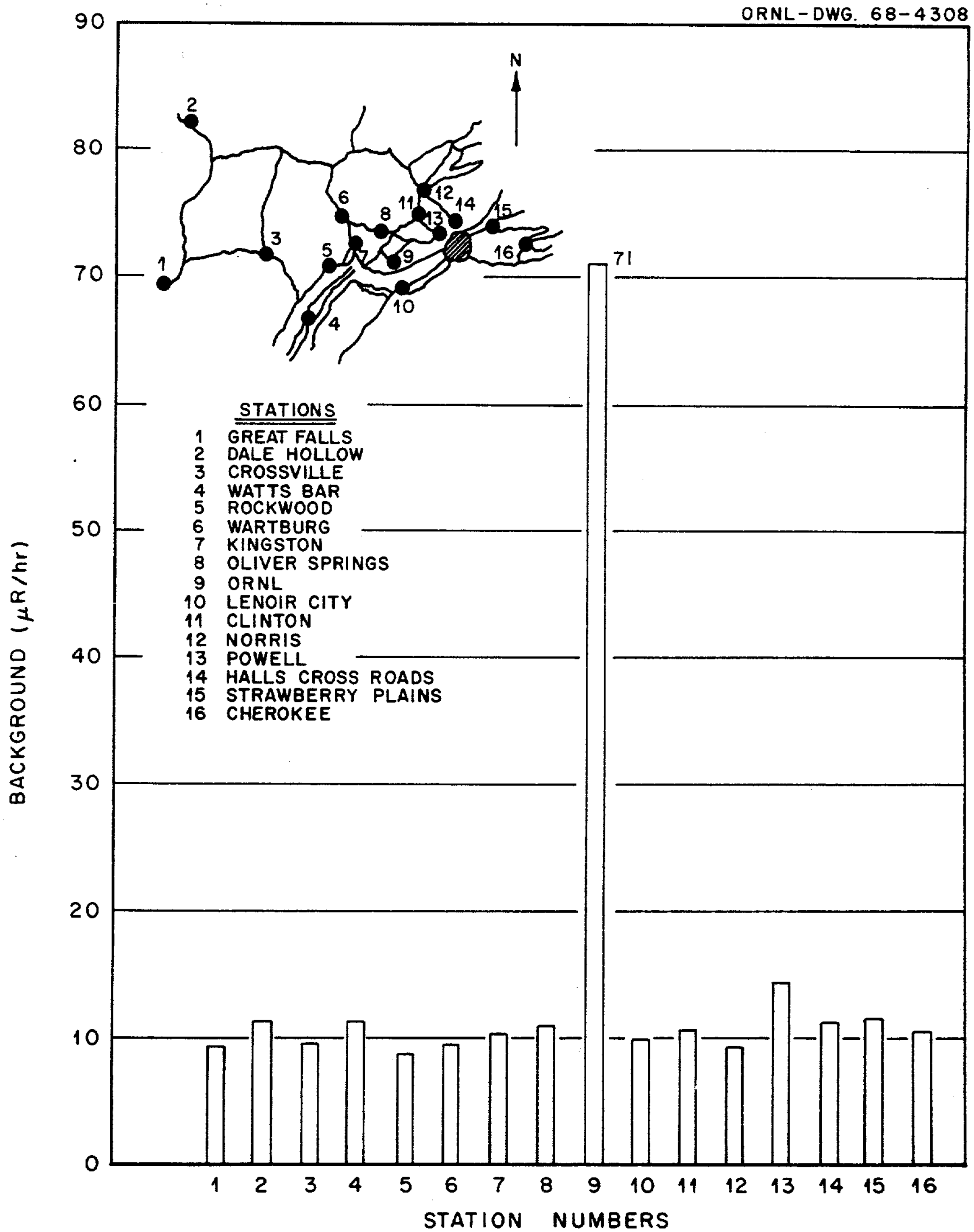

Fig. 5.14 Radiation Measurements Taken During 1967, 3 ft. Above the Ground Surface out to Distances of 75 Miles from ORNL.

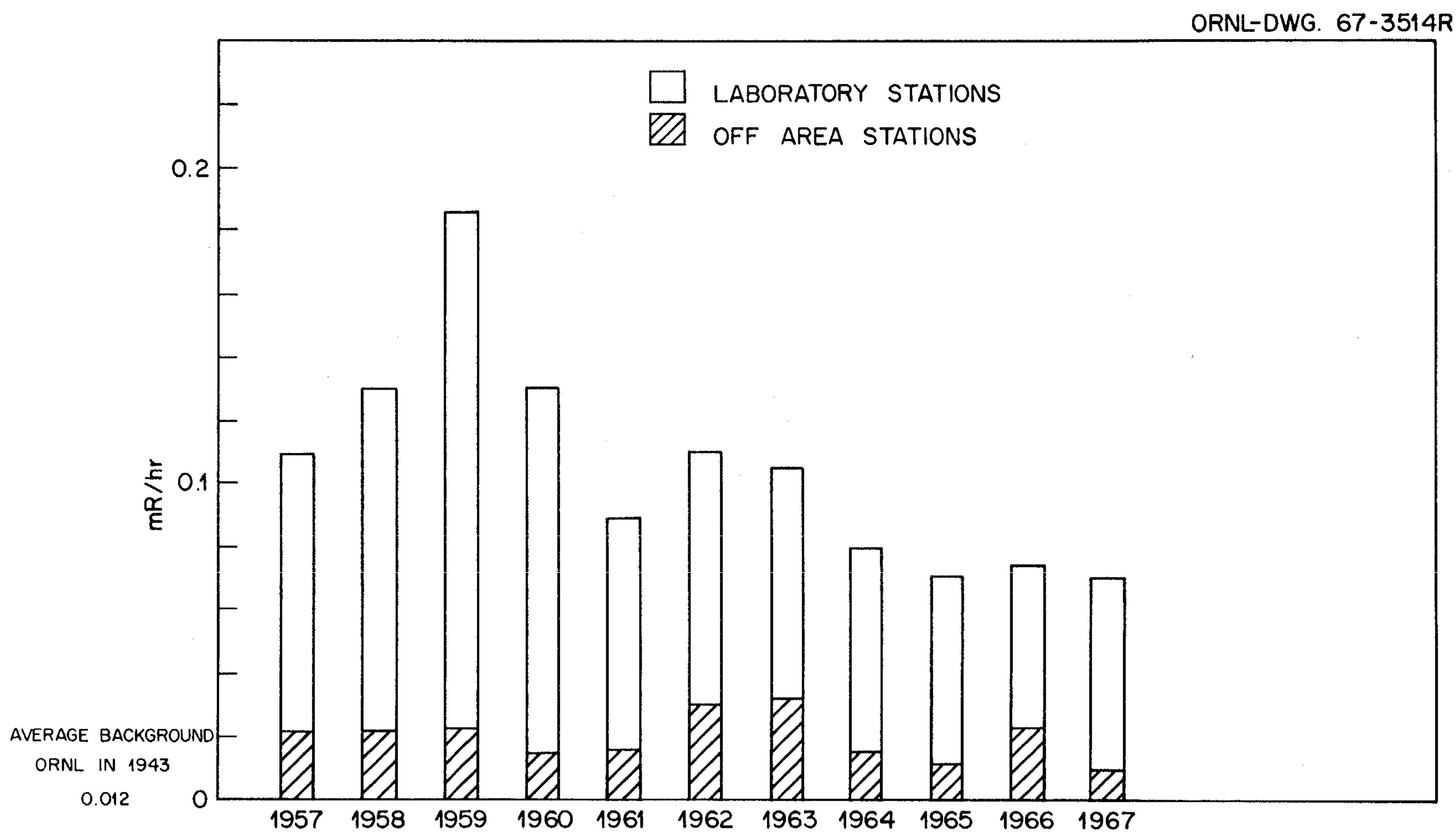

Fig. 5.15 Radiation Measurements Taken 3 ft. Above the Ground Surfaces at ORNL Compared with Like Measurements Taken Elsewhere within the AEC Controlled Area for the Years 1957-1967.

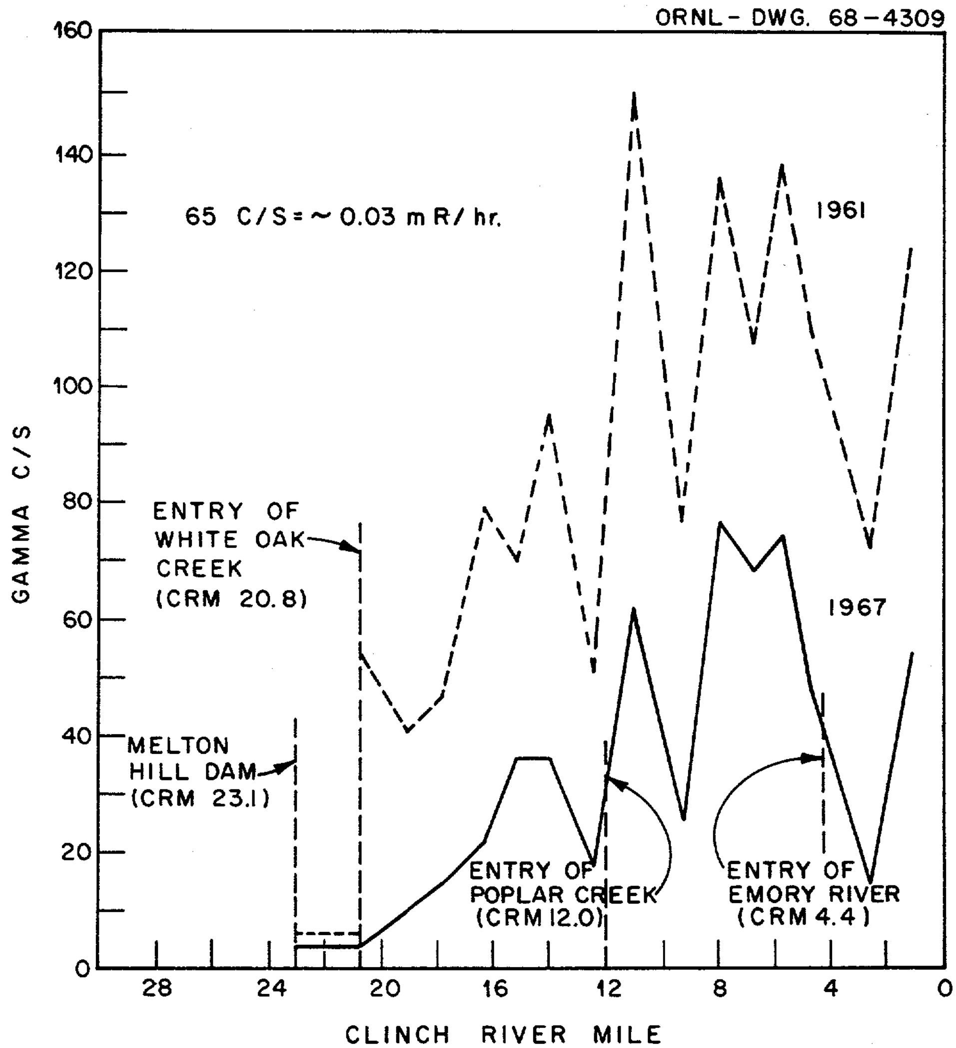

Fig. 5.16 Gamma Count at the Surface of Clinch River Silt.

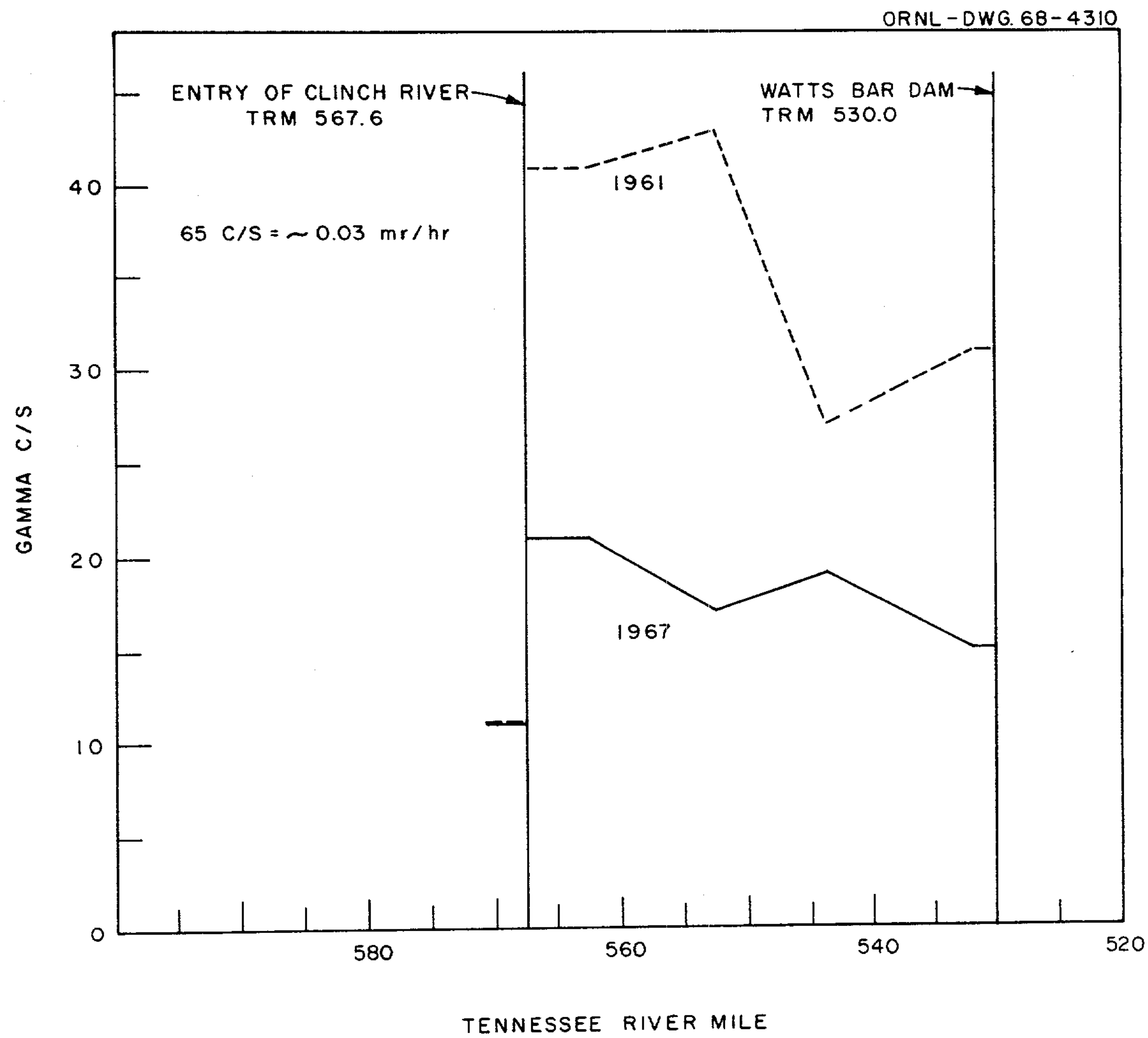

Fig. 5.17 Gamma Count at the Surface of Tennessee River Silt.

6.0 PERSONNEL MONITORING

It is the policy of the Oak Ridge National Laboratory to monitor the radiation exposure of all persons who enter Laboratory areas where there is a likelihood of radiation exposure. Dose analysis is accomplished mainly through the use of personnel meters, bio-assays, and in vivo counting (whole body counter) techniques.

6.1 Dose Analysis Summary, 1967

6.1.1 External Exposures - No employee received a whole body radiation dose which exceeded the maximum permissible levels recommended by the Federal Radiation Council (FRC). The highest whole body dose received by an employee was about 5.1 rem or 43 percent of the maximum permissible annual dose. The range of doses for persons using ORNL badge-meters is shown in Table 6.1.

As of December 31, 1967, no employee had a cumulative whole body dose which exceeded the recommended maximum permissible dose as based on the age proration formula 5(N-18) (Table 6.2). Only one employee had an average annual exposure rate that exceeded 5 rem per year of employment (Table 6.3).

The highest cumulative dose to the skin of the whole body received by an employee during 1967 was about 11 rem or 37 percent of the maximum permissible annual skin dose of 30 rem.

As of December 31, 1967, the highest cumulative dose of whole body radiation received by an employee was approximately 90 rem. This dose was accrued over an employment period of about 24 years and represented an average exposure of about 3.8 rem.

The highest cumulative hand exposure recorded during 1967 was about 25 rem or 33 percent of the recommended maximum permissible annual dose to the extremities.

The average of the ten highest whole body doses of ORNL employees for each of the years 1961 through 1967 are shown in Figure 6.1. The highest individual dose for each of those years is shown also.

The dose ranges versus the number of employees for each range for the years 1961 through 1967 are shown in Figure 6.2. Although the total number of employees increased slightly during the six-year period, the number of persons in the higher dose ranges has generally decreased.

The average annual dose to ORNL employees for the years 1961 through 1967 is the subject of Figure 6.3. This rather arbitrary quantity is obtained by dividing the sum of all doses for the year by the number of employees involved.

6.1.2 Internal Exposures - During 1967 there were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half a maximum permissible body burden.[6]

Three employees continued to have estimated body burdens of transuranic alpha emitters (mainly ^{239}Pu) of 35 to 40 percent of the recommended maximum permissible value.[7] The ICRP recommends, Publication 6, paragraph 86(9), individuals who exceed 50 percent of a maximum permissible body burden be placed on a work assignment where the potential for internal exposure is reduced.

6.2 External Dose Techniques

6.2.1 Film Meters - Film meters are issued to all persons who have access to ORNL facilities in which there is a likelihood of radiation exposures for which monitoring is required. Either an ORNL badge-meter (Figure 6.4) or a temporary pass-meter (Figure 6.5) may be used. Badge-meters are assigned to all ORNL employees, and to certain other persons who are authorized to enter ORNL facilities. Temporary pass-meters may be issued in lieu of badge-meters for short-term use.

NTA (nuclear track) film packets are included in all film meters. The NTA films are processed routinely if the badge-meter is assigned to an individual who normally works where there may be exposure to neutrons; otherwise the films would be processed only in the event of a nuclear accident.

Beta-gamma sensitive films from badge-meters issued to full-time employees are processed routinely each calendar quarter (or more frequently if necessary). Films used in other meters are processed as conditions of use may require. Films from meters issued to visitors are processed if there is a likelihood that a radiation exposure was incurred.

High-level radiation dosimetry components of the badge-meters (sulfur, gold, indium, and metaphosphate glass) are for use in the event that doses exceed the capability of the monitoring films.

For each ORNL division which had one or more employees who sustained a dose greater than 1 rem for the year, the number of employees so exposed are displayed in Figure 6.6. It may be noted that only ten (of 29) divisions had employees with doses greater than 1 rem, only eight had employees with doses greater than 2 rem, and only four had employees with doses greater than 3 rem.

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[6]Handbook 60 values are the basis for these determinations.

[7]AEC Manual Chapter 0502 requires an evaluation of the radiation exposure status of an employee when monitoring techniques indicate that a body burden equals or exceeds 50 percent of a maximum permissible limit.

6.2.2 Pocket Meters - Pocket meters (indirect reading, ionization chambers) are made available at all principal points of entry to ORNL premises. A pair of pocket meters is carried for the duration of a work shift by persons who work in an area where the potential for an exposure of 20 mR or more exists during the work shift. Pocket meter pairs are processed each day by Health Physics technicians and readings of 20 mR or more are reported daily to supervision. Pocket meters are used for a day-to-day record of integrated exposures.

Figure 6.7 is a display of the comparison between whole body doses as determined from film meters and the total recorded pocket meter readings for the ten highest whole body cases for the year 1967.

6.2.3 Hand Exposure Meters - Hand exposure meters (Figure 6.8) are film-loaded finger rings used to measure hand exposure. Hand exposure meters are issued to persons for use during operations where it is likely that the hand dose is such as to exceed 1 rem during the week. They are issued and collected by Radiation and Safety Surveys Section personnel who determine the need for this type of monitoring and arrange for a processing schedule.

6.2.4 Metering Resume - Shown in Table 6.4 are the quantities of personnel metering devices used and processed during 1967. The number of films processed is less than the number issued, because those which are issued for accident dosimetry only are not processed unless there was a likelihood of exposure.

6.3 Internal Dose Techniques

6.3.1 Bio-Assays - Urine and fecal samples are analyzed for the purpose of making internal dose determinations. The frequency of sampling and the type of radiochemical analysis performed is based upon each specific radioisotope and the exposure potential. Because of the small quantities of radioactive material in most samples, qualitative analyses are not feasible, and only quantitative analyses for predetermined isotopes are performed routinely.

In most cases bio-assay data require interpretation to determine the dose to the person; computer programs are used for evaluation of extensive data on urinary excretion of ^{239}Pu. An estimate of dose is made for all cases in which it appears that one-third of a body burden, averaged over a calendar year, may be exceeded.

6.3.2 Whole Body Counter - The whole body counter (an in vivo gamma spectrometer) may be used for determining internally deposited quantities of most of the gamma ray-emitting substances, and many of the more energetic beta-emitting substances. Thus, it provides a direct method of determining body burdens of those substances.

6.4 Records and Reports

Most records and reports are prepared by automatic data processing (ADP) techniques through the use of high-speed digital computer systems.

The IBM 7090, located at the Central Data Processing Facility (CDPF), provides routine weekly, quarterly, and annual reports involving external dose data. A typical weekly report is shown in Figure 6.9; a typical quarterly report is shown in Figure 6.10. A CDC 1604, operated by the ORNL Math Panel, is used to prepare the weekly pocket meter report (Figure 6.11) as well as the weekly, quarterly, and annual bio-assay reports. A sample of the Weekly Bio-Assay Sample Status Report is shown in Figure 6.12.

An annual report based on preliminary results of analysis by the whole body counter (IVGS) is prepared by the IBM 360 at CDPF.

A quarterly and an annual report of occupational injuries is processed by the IBM 360 at ORNL.

An individual external dose summary (Figure 6.13) is prepared annually by updating on the IBM 360 at CDPF.

Body burden estimates of ^{239}Pu are prepared in report form (usually quarterly) by use of the IBM 7090 at CDPF.

Permanent files are maintained at Health Physics and Safety Headquarters for each individual who is assigned an ORNL photo-badge-meter. An IBM card cross-indexing system is maintained at the principal monitoring stations for the purpose of expediting meter assignments. These IBM cards are compatible with the various computer programs and provide for the internal audit of all personnel monitoring record data.

Copies of the ADP reports, both temporary and final, are maintained for both the internal and external dose programs. Data used in the ADP program are stored on computer quality magnetic tapes. Data pertinent to the work of the dosimetry groups and information used in the non-ADP reports are maintained in record form by the Dose Data Group.

6.5 Program Developments

During 1967 the computer program for preliminary reports of IVGS analysis was modified for preparation on the IBM 360.

The program for revising and increasing the utility of bio-assay results from prior years was completed. Computer reports for the calendar years 1951-1965 have been prepared.

Radiophotoluminescent (RPL) glass dosimeters, Toshiba FD-P6, and a reading device for analyzing them have been obtained. The RPL dosimeters will be installed in most ORNL badge-meters during 1968. They will be used in the event of a faulty or damaged dosimeter film from the badge.

Table 6.1 Dose Data Summary for Laboratory Population Involving Exposure to Whole Body Radiation - 1967

Group	Number of Rem Doses in Each Range							Total
	0-1	1-2	2-3	3-4	4-5	5-6	6 up	
ORNL Employees	5827	140	54	17	3	1	0	6042
ORNL-Badged Non-Employees	1088	5	0	0	0	0	0	1093
TOTAL	6915	145	54	17	3	1	0	7135

Table 6.2 Average Rem Per Year Since Age 18 - 1967

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	6034	8	0	0	6042

Table 6.3 Average Rem Per Year of Employment at ORNL - 1967

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	6014	27	1	0	6042

Table 6.4 Personnel Meter Services

	1965	1966	1967
A. Pocket Meter Usage			
1. Number of Pairs Used			
ORNL	140,088	156,676	150,748
CPFF	19,656	17,108	19,344
Total	159,744	173,784	170,092
2. Average Number of Users per Quarter			
ORNL	1,262	1,372	1,408
CPFF	256	213	252
Total	1,518	1,585	1,660
B. Film Usage			
1. Films Used in Photo-Badge-Meters			
Beta-Gamma	21,810	21,760	20,800
NTA	10,830	10,670	10,300
2. Films Used in Temporary Meters			
Beta-Gamma	7,720	7,790	4,930
NTA	2,500	2,520	1,600
C. Films Processed for Monitoring Data			
1. Beta-Gamma	22,080	22,190	21,150
2. NTA	1,690	2,470	1,580
3. Hand Meter	1,610	1,940	2,490

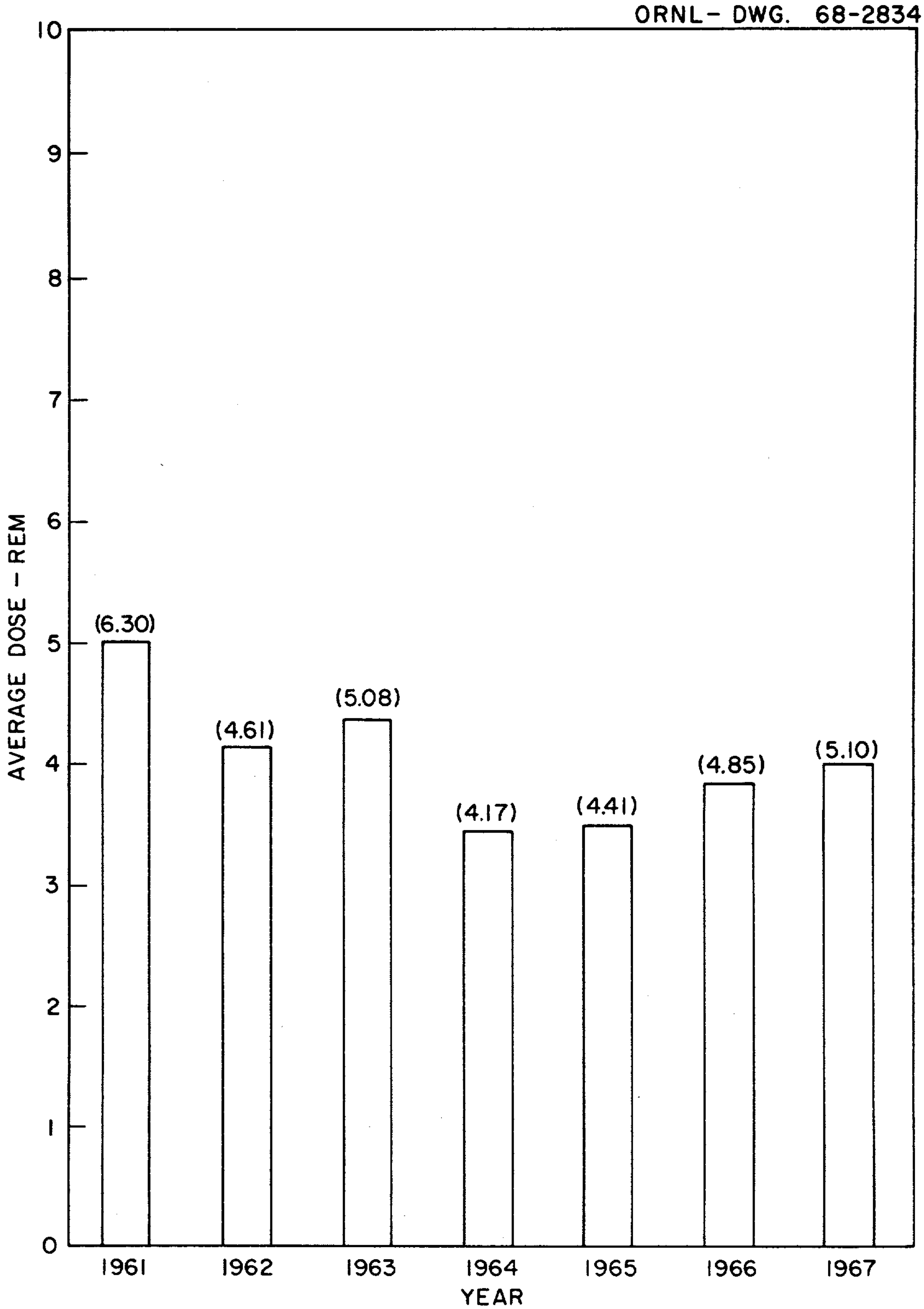

Fig. 6.1 Average of the Ten Highest Annual Whole Body Doses by Year (The Highest Individual Dose Shown in Parentheses).

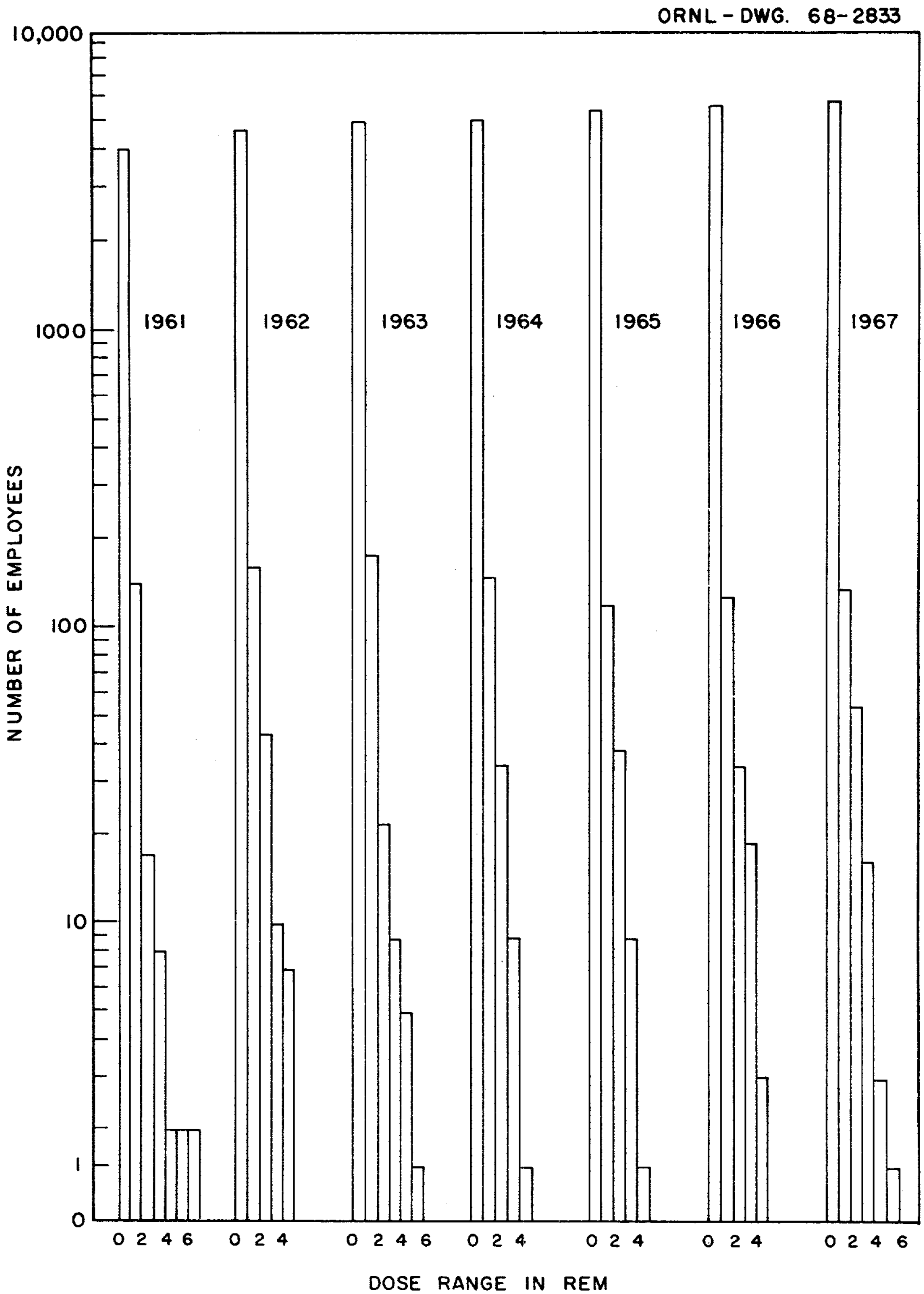

Fig. 6.2 Whole Body Radiation Dose Range for Employees 1961-1967.

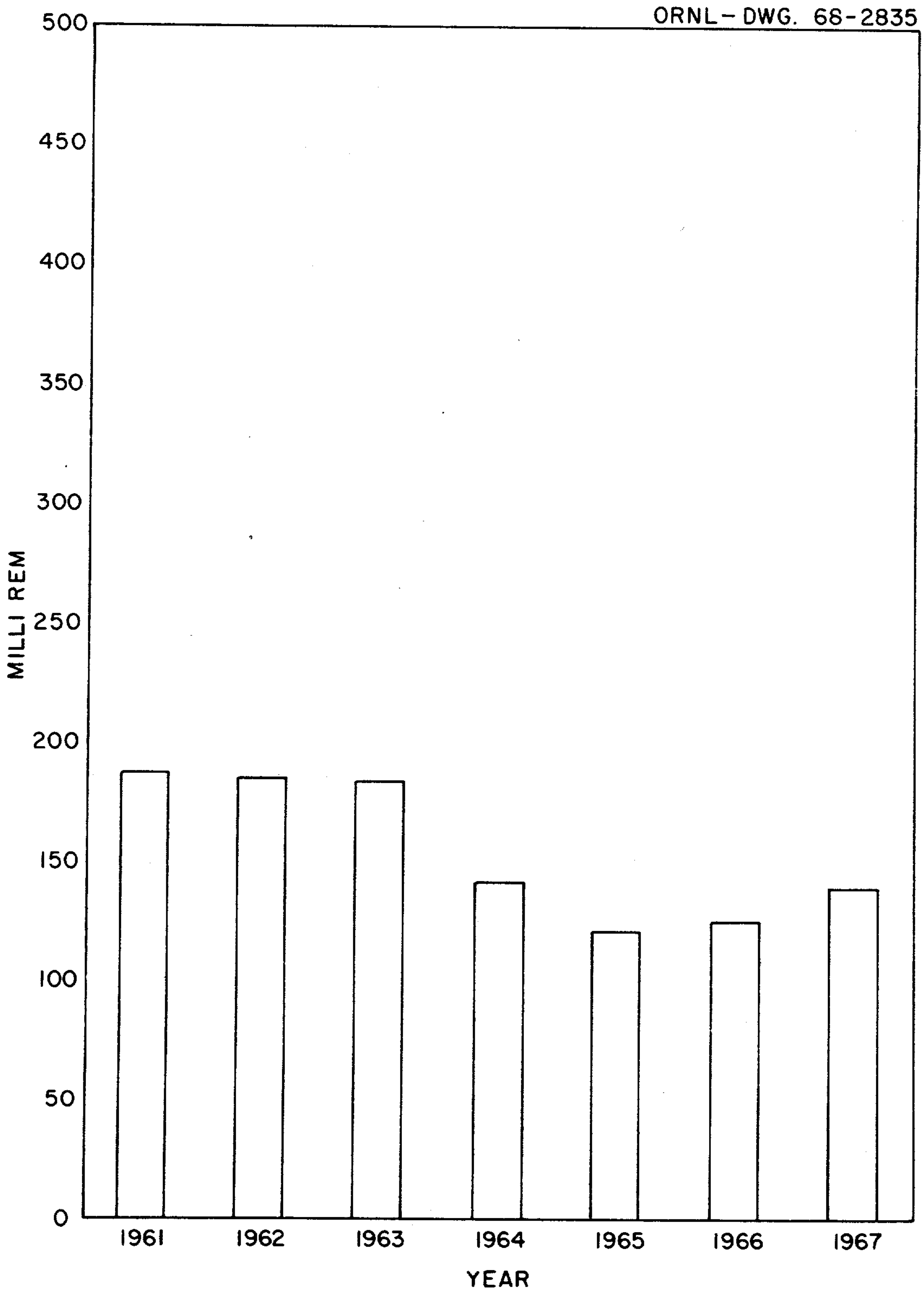

Fig. 6.3 Average Annual Whole Body Dose to the Average ORNL Employee.

ORNL–LR–DWG 50112R2

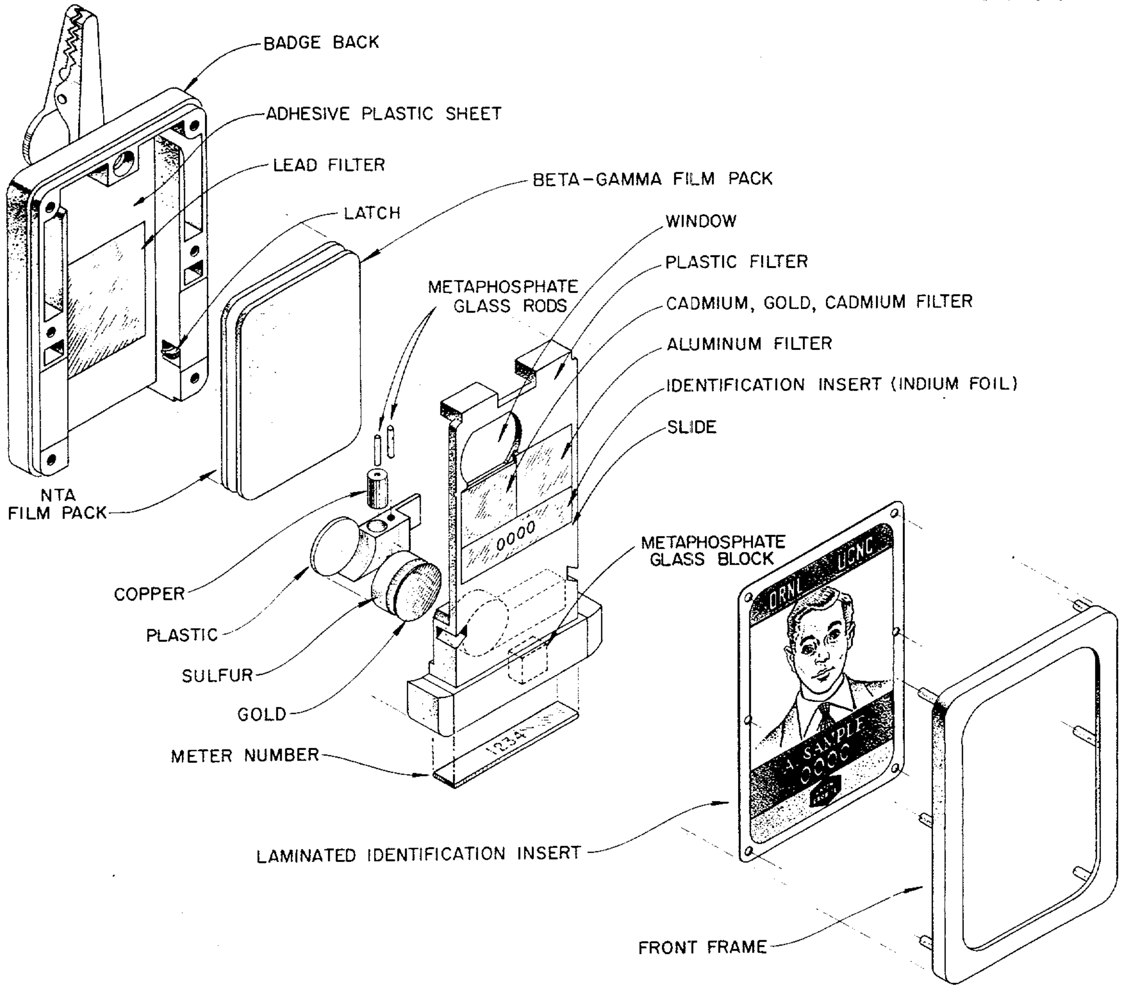

Fig. 6.4 ORNL Badge-Meter, Model II.

PHOTO 89910

Fig. 6.5 Typical Temporary Security Passes Equipped with Monitoring Film.

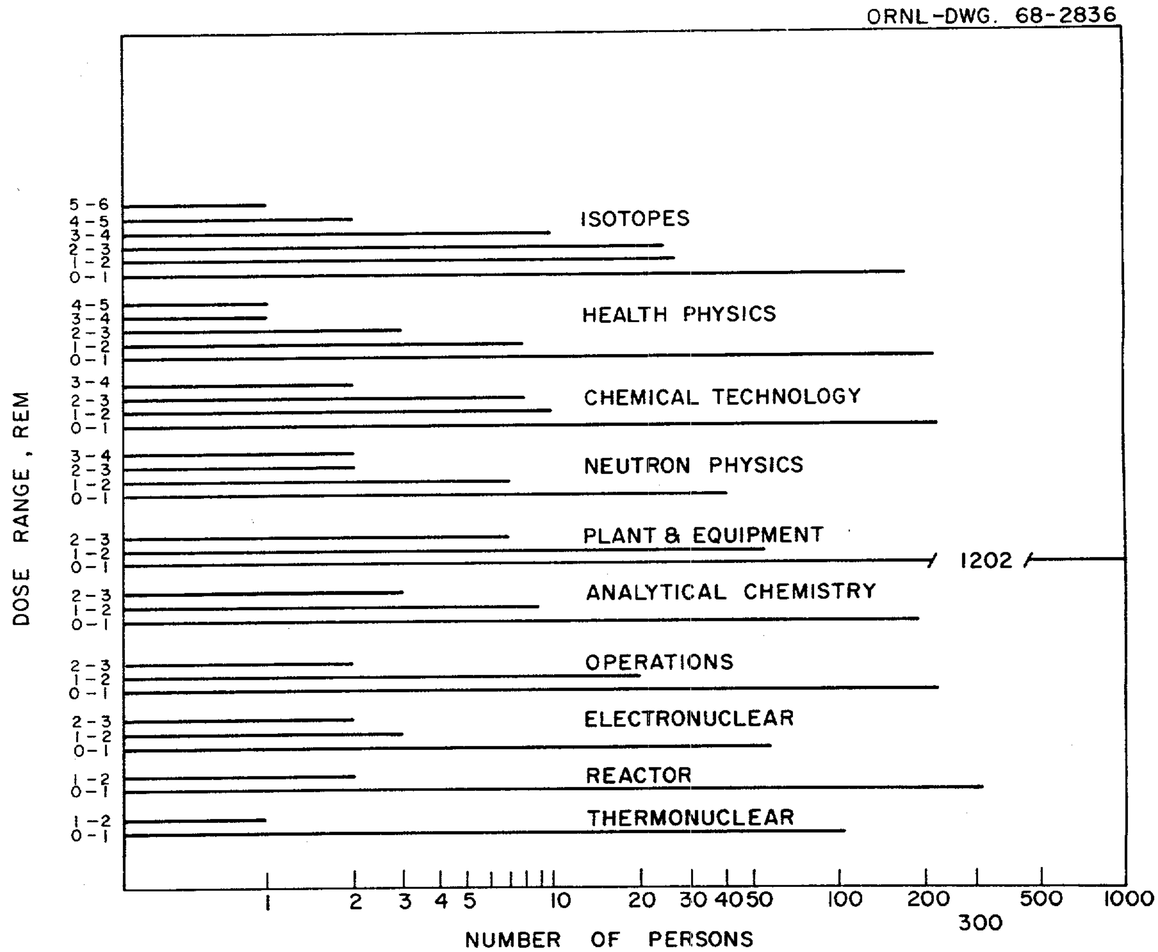

Fig. 6.6 Personnel Dose (Whole Body) by ORNL Division Having One or More Doses, One Rem or Greater, in 1967.

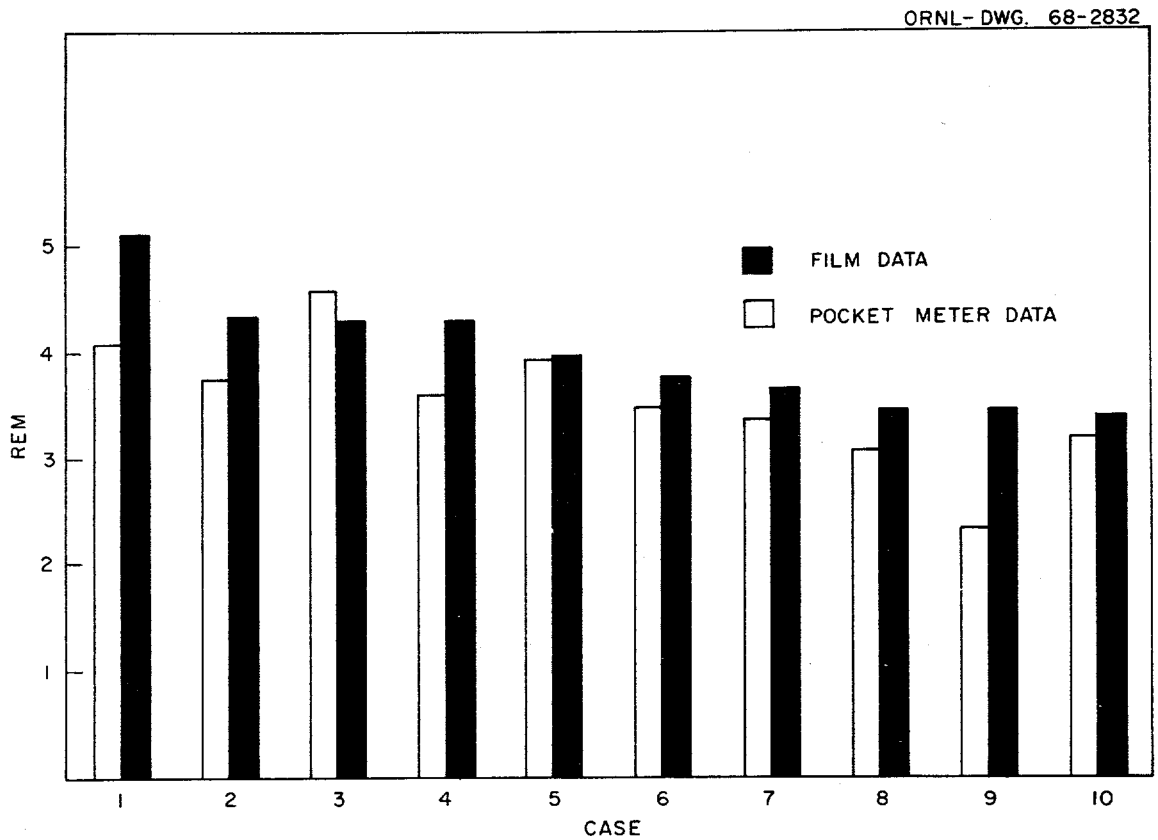

Fig. 6.7 The Ten Highest Whole Body Radiation Dose Cases Compared with Concurrent Pocket Meter Totals for 1967.

ORNL-DWG. 66-1814

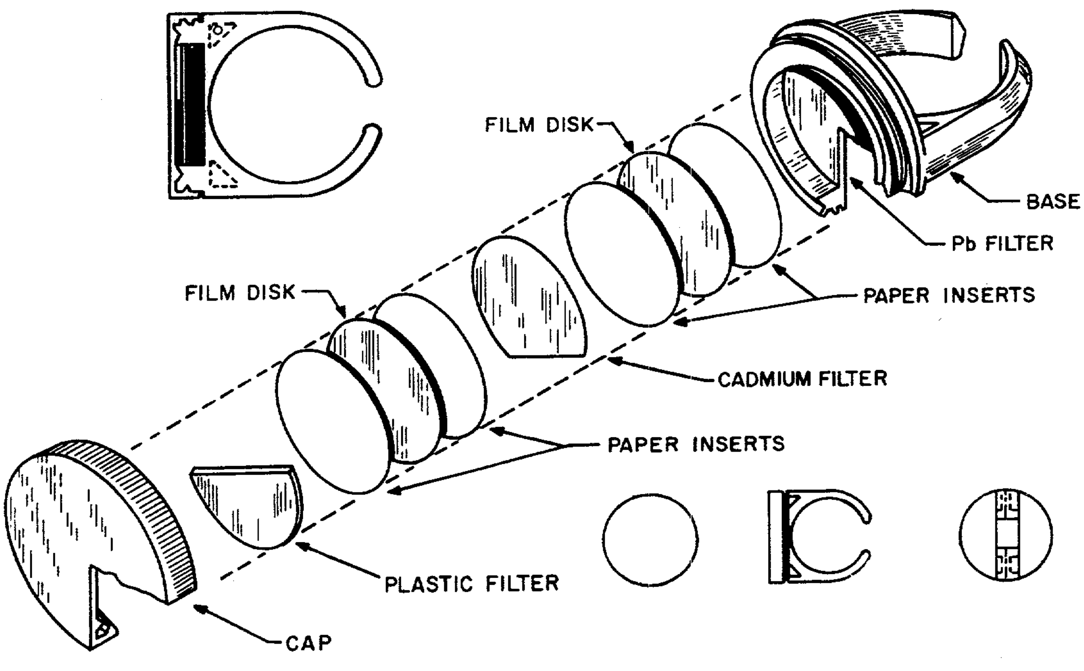

Fig. 6.8 Details of the ORNL Hand Exposure Meter.

ORNL DWG. 64-11676

Name	ID Number	Symbol	Dosimetry Dates		Meter Dose	
			Wk-Yr	Qtr-Yr	DS	DC
Last Name, Initials	PR. No.	PF	35-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	31-63	3-63	0.120	0.090
Last Name, Initials	PR. No.	PF	30-63	3-63	0.030	0.000
Last Name, Initials	PR. No.	PF	36-63	3-63	0.070	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	36-63	3-36	0.370	0.310
Last Name, Initials	PR. No.	PF	32-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	33-63	3-63	0.040	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.260	0.130
Last Name, Initials	PR. No.	PF	35-63	3-63	0.040	0.010

Fig. 6.9 Typical ORNL Film Monitoring Data.

DWG 64-11675

HEALTH PHYSICS DIVISION
DEPARTMENT 3193
RADIATION SURVEY

Name	ID Number	Symbol	Date Wk-Yr	REM DS	REM DC	REM This Qtr DS	REM This Qtr DC	REM This Yr DS	REM This Yr DC	Total REM DC	A	DC/A
----	-----	PF PN	39-63 39-63	0.860 0.000	0.630 0.000	0.860	0.630	1.68	1.32	35.59	18	2.02
----	-----	PF	39-63	0.340	0.240	0.340	0.240	0.34	0.24	0.24	1	0.80
----	-----	PF	39-63	0.020	0.010	0.020	0.010	0.02	0.01	5.21	14	0.38
----	-----	PF	39-63	0.070	0.040	0.070	0.040	0.30	0.19	18.38	16	1.19
----	-----	PF	39-63	0.390	0.310	0.390	0.310	1.40	1.14	2.74	20	0.14
----	-----	PF	39-63	0.350	0.150	0.350	0.150	0.77	0.49	9.60	17	0.56
----	-----	PEL PF PN	27-63 39-63 39-63	0.010 0.140 0.000	0.010 0.110 0.000	0.150	0.120	0.27	0.24	5.55	6	1.09
----	-----	PF	39-63	0.400	0.200	0.400	0.200	0.73	0.45	7.43	12	0.64
----	-----	PF	39-63	0.180	0.150	0.180	0.150	0.60	0.49	8.43	7	1.34
----	-----	PF PN	39-63 39-63	0.330 0.030	0.110 0.030	0.360	0.140	0.81	0.34	3.00	13	0.24
----	-----	PF PN	39-63 39-63	0.180 0.000	0.080 0.000	0.180	0.080	0.51	0.33	29.82	18	1.68
----	-----	PF PN	39-63 39-63	0.320 0.000	0.270 0.000	0.320	0.270	1.14	0.98	22.76	13	1.76
----	-----	PF	39-63	0.420	0.290	0.420	0.290	1.85	1.11	15.86	16	1.04
----	-----	PF	39-63	0.320	0.140	0.320	0.140	0.67	0.46	8.96	11	0.84
----	-----	PF	39-63	0.390	0.210	0.390	0.210	1.21	0.72	33.62	18	1.87

Fig. 6.10 Typical ORNL Personnel Radiation Exposure Record.

ORNL DWG. 66-4519

DEPT XXXX HP WK 52

NAME	PR NO	DC	S	M	T	W	T	F	S	WK	QTR	F	SMB
---------------	-----			0	5	10	0			15	270	62	
---------------	-----	660		10	25		120			155	1140	57	DWQ
---------------	-----										35	11	
---------------	-----										10	1	
---------------	-----			10	10	0	10			30	220	41	
---------------	-----			10						10	115	62	
---------------	-----										5	20	
---------------	-----		0	10	10		20			40	560	54	D Q
---------------	-----										195	22	
---------------	-----											1	
---------------	-----		125	0		0				125	260	60	DW
---------------	-----			30	0	5	0			35	415	61	D
						ENTRIES		+		D	W	Q	COUNT
						23				5	2	2	12

Fig. 6.11 Typical Pocket Meter Weekly Report.

ORNL DWG. 66-4520

RESULTS THIS REPORT 12-20-65

Div. Code	Name	PR NO	HP AREA Number	Type Analysis	Receipt Date	Type Sample	Sample Priority	d/m/Sample	d/m/24 hrs
HP	----------	-----	3550	GAO	12-16-65	U	3		0
HP	----------	-----	3019	PUO	12-12-65	U	3		0
HP	----------	-----	3019	PUO	12-16-65	U	3		0
HP	----------	-----	3019	SRO	12-12-65	U	3		0
Div. Total 4									

Fig. 6.12 Typical Weekly Bio-Assay Sample Status Report.

ORNL-DWG. 67-2638

Page 1 of 2

ORNL EXTERNAL RADIATION EXPOSURE RECORD

Name - Employee AN

		Symbol	Definition
I.D. Number	5782	DC	Cumulative recorded total rem to whole body since activation date.
S.S. Number	221-16-0038		
Birth Date	6/17/28	DO	Dose data other than that reported herein Yes (3)
Activation Date	1/16/48		

Year	QTR	Rem for Qtr Skin	Rem for Qtr Body	Rem for Year Skin	Rem for Year Body	Rem DC
				DC Prior to 1961		22.13
1961	1	.26	.19			
	2	.20	.16			
	3	.29	.12			
	4	.44	.36			
Total				1.30	.83	22.96
1962	1	.33	.30			
	2*	.56	.48			
	3*	.69	.54			
	4	.59	.51			
Total				2.17	1.83	24.79
1963	1	.61	.50			
	2	.53	.43			
	3	.78	.43			
	4	.03	.03			
Total				1.95	1.39	26.18
1964	1	.04	.03			
	2	.02	.01			
	3	.02	.01			
	4	.09	.04			
Total				.17	.09	26.27
1965	1	.25	.12			
	2	.40	.22			
	3	.48	.28			
	4	.41	.21			
Total				1.54	.83	27.10

*See last page for termination and/or reinstatement dates.

Fig. 6.13 Typical Individual External Dose Summary.

7.0 LABORATORY OPERATIONS MONITORING

Radiation incidents are classified according to a severity index system developed over the past several years.[8] The method serves to index unusual occurrences according to degree of severity and permits a system of analysis regarding Health Physics and Safety practices among Laboratory operations. This report summarizes the unusual occurrence frequency rate and discusses some of the problems encountered among Laboratory facilities.

7.1 Unusual Occurrences

During 1967 there were 16 unusual occurrences recorded which represents a decrease of approximately 27 percent from the number reported for 1966 (Table 7.1). The number for 1967, 16, is approximately 47 percent below the five-year average of 30 for the years 1963 through 1967. The frequency rate of unusual occurrences among the Laboratory divisions involved (Table 7.2) is known to vary in relationship to the quantity of radioactive material handled, the number of radiation workers involved, and the radiation hazard potential associated with a particular operation or facility.

Eleven of the incidents reported during 1967 involved primarily area contamination that was handled by the regular work staff without appreciable production or program loss. Three occurrences involved personnel contamination and two were concerned with external exposures requiring minor work restrictions. There were no overexposure cases reported during the year.

7.2 Radiation Surveys

During 1967 Radiation and Safety Surveys personnel assisted the operating groups in keeping the contamination, air concentration, and personnel exposure levels well below the established maximum permissible limits. Through seminars, safety meetings and informal discussions with supervision, they assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory. The following is a brief description of some of the problems and methods of solution.

7.2.1 Health Physics Coverage at the Radio Isotopic Sand Tracer Project - A representative of the Radiation and Safety Surveys Section acted as project health physicist at the Radio Isotopic Sand Tracer Tests, conducted by the Technical Services Group of the Isotopes Division for the U. S. Corps of Engineers, at Vandenberg Air Force Base, California and Point Conception, California. The tests involved placing radioactive sand (^{133}Xe encapsulated in sand) offshore in the ocean and tracing its movement along the ocean floor by use of a specially designed radiation detection system. The Health Physics representative, in providing on-the-job

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[8] See Applied Health Physics Annual Report for 1963, ORNL-3665, pp. 14-15.

surveillance, served as custodian of radioactive material as well as assuring that all Federal and State regulations pertaining to the handling of the material were followed. Also, a short Health Physics training course was conducted for the Corps of Engineers personnel involved in the tests. The tests were completed without significant contamination or exposure problems.

7.2.2 Health Physics and Safety Assistance during the Disposal of Liquid Waste at the Shale Fracturing Site - Continuous monitoring was provided Operations Division personnel during the injection of $\sim 8.7 \times 10^4$ curies of activity in concentrated liquid waste at the Shale Fracturing disposal facility. During one of the injection operations the internal system of a standby pumper became contaminated due to the backup of liquid solution. However, decontamination efforts were successful in reducing the activity to acceptable levels prior to the release of the equipment for other work. Radiation exposure to personnel was kept well below the permissible limits and, with the exception noted above, no significant contamination problems were encountered.

7.2.3 Health Physics and Safety Assistance during Production of Material in Transuranic Facility, Building 7920 - The TRU facility established several "firsts", within the atomic energy program, relating to the quantity of certain materials produced during the year. Approximately one milligram of ^{252}Cf and microgram quantities of $^{253-254}$Es and ^{249}Bk were prepared and shipped during the period. Gram quantities of ^{243}Am and ^{244}Cm were processed in the cells, and gloved boxes in the analytical laboratories also handled comparable amounts of ^{238}Pu and $^{241-243}$Am.

The quantity and various energies of neutrons emanating from some of these materials created additional shielding and monitoring problems not normally associated with other facilities at the Laboratory. In addition to this special problem, the highly contaminated processing cells were entered on several occasions to repair or modify equipment. Although contamination levels in certain areas of the cells were $> 10^6$ d/m/100 cm^2 α the work was performed without significant personnel contamination or release of material outside the established C-Zones. Radiation and Safety Surveys personnel participated in helping to plan procedures for controlling these problems as well as providing on-the-job surveillance.

7.2.4 TRUST (Thorium Reactor Uranium Storage Tank) Facility Construction, Building 3019 - Facilities for long-term storage of 1200 kg of uranium (75 percent ^{235}U, 11 percent ^{233}U concentration of 120 ppm) as uranyl nitrate solution, are under construction at Building 3019. When completed, the facility will include a gloved box in the penthouse to receive the bottled solution; a cubicle in cell 4 for transferring and sampling the solution; and a 5000 gal storage tank in the thorium decay pit south of Building 3019.

Specifications for storage of this material required renovation of the decay pit and the storage tank—both alpha contaminated through use in prior years. The tank was thoroughly flushed in position in the pit and then removed to the burial ground where additional decontamination enabled personnel wearing masks, coveralls, gloves and shoe covers to carry out the necessary inspection and repair operations. Through decontamination and paint bonding much of the pit renovation required no protective apparel or a very minimum of protective apparel.

The old Kilorod sol gel cubicle in cell 4 was renovated to receive equipment for sampling the uranium solution and transferring it to the storage tank in the decay pit. Decontamination of this cubicle removed gross alpha contamination so that air supplied plastic suits were no longer required but rather masks and a double suiting of coveralls, gloves, and shoe covers sufficed. Due to gamma radiation from ^{233}U stored in the adjacent cell wall, rigid control was necessary to maintain external exposure to construction personnel within permissible limits.

Full cooperation was received from all groups concerned in performing the above operations with minimal exposure to personnel and only minor spread of contamination.

7.2.5 Tie-In of New Ventilation Filter House at Building 3517 - A new cell ventilation filter house containing three banks of CWS filters in parallel was tied into the existing system at Building 3517 during the month of December. The cell ventilation duct (690 ft. of 4 ft. diameter concrete pipe) was washed down from the 3517 filter pit to the central off-gas facility. Following the successful decontamination effort minor adjustments and testing were made and the filters placed in service.

Radiation and Safety Surveys personnel participated in the planning of the above operation and provided on-the-job surveillance. With adequate pre-planning and adherence to radiation and contamination control procedures the work was completed with minimal exposure to personnel and spread of contamination.

7.2.6 Construction of a Pu Sol Gel Pilot Plant in Cell 4, Building 3019 - Health Physics assistance was provided during the renovation of the Kilorod Facility inspection cubicle and during installation of new equipment for Sol Gel plutonium operations. After removal of gross alpha contamination from the cubicle surfaces the remaining contamination was bonded by paint. A series of air samples indicated the paint bond was effective in maintaining airborne contamination well below permissible levels. Thus, new equipment was installed with a minimum of protective apparel and without the necessity of respiratory protection, resulting in greatly improved worker efficiency.

7.2.7 Installation of a New Process Off-Gas Filter System for the Radiochemical Pilot Plant (Building 3019) - A new process off-gas filter system was constructed in a new concrete block structure (Building 3121) at the southeast corner of Building 3019. This new filter system provides improved efficiency; a parallel, filtered, by-pass; bagging technique for filter replacement; and a critically safe condensate collection system.

Removal of the old filter system and transfer to the new system presented some unique problems. The old filter system, grossly alpha contaminated and containing pockets of acidic condensate, was situated near unrestricted vehicular and pedestrian traffic routes, outside building containment. In addition, the old system contained no by-pass which would enable tie-in of the new system without interruption of essential off-gas to primary containment gloved boxes and vessels. In-flow of air into the off-gas system also had to be minimized in order to assure sufficient off-gas services for other ORNL facilities.

A temporary by-pass was installed which enabled continuation of off-gas service to Building 3019 during the switch over. The old contaminated off-gas pipe sections were removed in a special sequence so that they were under a negative pressure until they were adequately contained for transfer to the burial ground. In-flow was minimized by simultaneously inserting blanks when a pipe section was opened. With rigid control throughout and careful pre-planning of each phase, this job was completed without any significant release of radioactive materials and without any serious reductions of off-gas service to any ORNL facilities.

7.2.8 Initial Operations of the TSF-SNAP Reactor - Health Physics personnel provided the radiation monitoring surveillance necessary during the assembly, critical experiments, and initial operations of the TSF-SNAP reactor at the Tower Shielding Facility.

The fuel elements (SNAPTRAN) were free of contamination on arrival so that the handling of them was simplified throughout the dry critical experiment phases.

The NaK loading was completed in mid-June and experimental operations were begun shortly thereafter. No difficulties or significant exposures had been experienced to the end of the year.

7.2.9 Renovation of Building 3003 and Installation of a 2 MeV Van de Graaff Machine - Constant Health Physics surveillance was provided during the renovation of Building 3003 prior to the installation of the 2 MeV Van de Graaff machine. This work consisted of assuring radiation and contamination control during the removal of the 50,000 CFM fan and associated duct work located in Cell I. The work was accomplished without significant external or internal exposure to personnel.

Following installation of the machine numerous radiation surveys were conducted by Health Physics personnel during the period that the equipment was being checked out prior to acceptance by ORNL. These surveys indicated that additional shielding would be required before the machine could be operated at its maximum potential without exceeding permissible radiation levels in certain occupied areas. This problem was solved by enclosing the target area within an eight-inch Barytes concrete wall. In addition to the shielding problem, much effort was expended in conjunction with Solid State Division personnel on the overall safety aspects of the operation with particular emphasis on instrumentation and interlock systems.

7.2.10 Annual Survey of X-Ray Equipment - The annual survey of X-ray producing devices at X-10 and the ORNL portion of the Y-12 operation was completed during December, 1967. There are 82 X-ray producing devices currently registered. The survey revealed only a few minor exceptions to operational practices as recommended in Procedure 2.8 of the Health Physics Manual, and these are in the process of being corrected.

7.2.11 Health Physics and Safety Activities in the Transuranium Research Laboratory during 1967 - The Health Physics and Safety staff of the TRL facility participated in pre-operational safety reviews conducted by the ORNL Radiochemical Plants Committee on March 7, 1967, and by members of the Research and Development Division, AEC-ORO on March 14-15, 1967. The TRL Safety Analysis Report, prepared by the Health Physics and Safety staff, was used as the basis for these reviews. Operational approval was granted April 27, 1967, and shortly thereafter the first transuranic nuclides were introduced into the facility.

During the year the Health Physics and Safety staff were involved in seven primary areas of activity: (1) Building Containment Operations, (2) Safety Review and Planning of Specific TRL Experiments, (3) Technological and Protective Assistance during Experiments, (4) Health Physics and Safety Audits and Inspections, (5) Consultation with Other AEC Sites, (6) Safety Development, and (7) Conducting Training Program.

7.3 Laundry Monitoring

A total of 891,105 articles of wearing apparel was monitored at the laundry during 1967. Less than one percent of the items monitored were found to be contaminated.

Of the 416,592 khaki garments monitored during the year, only 60 were found contaminated. This is a decrease of about 39 percent from last year.

A total of 14,684 full-face respirators were cleaned and monitored during the year. Of this number, 909 required additional decontamination measures prior to being placed back in service.

Table 7.1 Unusual Occurrences Summarized for the 5-Year Period Ending with 1967

	1963	1964	1965	1966	1967
Number of Unusual Occurrences Recorded..........	43	29	41	22	16
A. Number of incidents of minor consequence involving personnel exposure below MPE limits and requiring little or no clean-up effort..................................	11	14	11	8	5
B. Number of incidents involving personnel exposure above MPE limits and/or resulting in special cleanup effort as the result of contamination..............................	32	15	30	14	11
1. Personnel Exposures.....................	4	9	12	8	5
a. Nonreportable overexposures with minor work restrictions imposed......	3	9	11	8	5
b. Reportable overexposures with work restrictions imposed.................	1	0	1	0	0
2. Contamination of Work Area..............	32	15	28	14	11
a. Contamination that could be handled by the regular work staff with no appreciable departmental program loss................................	30	14	27	12	11
b. Required interdepartmental assistance with minor departmental program loss........................	2	1	1	2	0
c. Resulted in halting or temporarily deterring parts of the Laboratory program..............................	0	0	0	0	0

Table 7.2 Unusual Occurrence Frequency Rate within the Divisions for the 5-Year Period Ending with 1967

Division	No. of Unusual Occurrences 1963	1964	1965	1966	1967	5-Year Total	Percent Lab. Total (5-Year Period)
Analytical Chemistry	9	3	6	1	3	22	14.5
Biology	2		1			3	1.9
Chemical Technology	11	3	8	3	4	29	19.2
Plant and Equipment	1	2	2	2		7	4.6
Inspection Engineering		1				1	.6
Electronuclear Research			1	1		2	1.3
Health Physics	1*	1	2			4	2.6
Instrumentation and Controls		1				1	.6
Isotopes	5	12	10	8	4	39	25.8
Metals and Ceramics	1				1	2	1.3
Neutron Physics	2					2	1.3
Operations	9*	3	8	4		24	15.8
Physics	3	3	2	1	1	10	6.6
Reactor				2	3	5	3.3
Reactor Chemistry			1			1	.6
TOTALS	43	29	41	22	16	151	100.0

*Shared responsibility with another division for one unusual occurrence.

8.0 INDUSTRIAL SAFETY

During the year emphasis was placed on the wearing of eye protection in posted zones or areas. This perhaps in some measure accounted for the 22 percent decrease in eye injuries for 1967 as compared to 1966.

8.1 Accident Analyses

The Disabling Injury Frequency Rate for 1967 was 0.50. The average frequency rate for the previous five years, 1962-1966, was 1.4. The disabling injury history of the Laboratory for the five-year period 1963 through 1967 is shown in Table 8.1. The disabling injury frequency rates since the inception of Union Carbide as the contractor at ORNL are shown in Figure 8.1.

There were eight divisions which did not have a Serious or disabling injury during 1967. There are 14 divisions which have accumulated 1,000,000 or more hours since the last disabling injury. The Serious Injury, disabling injury, and exposure-hour data for ORNL divisions are shown in Table 8.2.

Table 8.3 includes injury data for the four facilities—ORNL, Paducah, Y-12 and ORGDP. The frequency rates for disabling injuries for all four Carbide facilities decreased in 1967 as compared with 1966. The frequency rates for Serious Injuries decreased at three of the facilities, including ORNL. Serious Injuries at ORNL decreased from 93 in 1966 to 89 in 1967. The frequency rates for disabling injuries and Serious Injuries at ORNL for the past five years, 1963-1967, are shown graphically in Figure 8.2.

There were 1,557 injuries (includes first air, Serious Injuries, and disabling injuries) reported during 1967. Figures 8.3, 8.4, and 8.5 show injury data according to type of accident, the nature of the injury, and the part of body injured.

8.2 Analyses of Disabling Injuries

The following is a brief analyses of the four disabling injuries experienced at ORNL during 1967.

Date of Injury - 3/13/67

A technician backed down a slight ramp pulling a portable lift. When the front casters dropped off the end of the ramp (2" drop), the lift toppled forward. The technician was pinned underneath and sustained a fractured vertebrae.

Date of Injury - 3/21/67

A millwright stood outside the handrail on the edge of a concrete stoop (his back to the rail) while using a portable electric drill. His foot slipped and he fell to the pavement, sustaining a concussion.

Date of Injury - 6/17/67

A welder stepped from a scaffold board to a concrete pier two feet away. His bifocal glasses caused him to miss his step and fall three feet to the ground. His back struck against the pier, causing contusions and abrasions.

Date of Injury - 10/3/67

A mail clerk struck his heel with a four-wheel delivery cart which he was pulling. The small laceration, despite prompt medical treatment, became infected and involved the lymph glands behind his knee.

8.3 Safety Award Periods - 1967

December 17, 1966 - March 12, 1967	
1,771,556 hours -	$2.00
March 22, 1967 - June 16, 1967	
1,968,901 hours -	$2.00
June 18, 1967 - October 2, 1967	
2,372,887 hours -	$2.00
Total Award Value	$6.00

Table 8.1 Disabling Injury History

	1962	1963	1964	1965	1966	1967
Number of Injuries	10	11	8	18	4	4
Labor Hours (Millions)	6.9	7.1	7.5	7.7	7.8	8.0
Frequency Rate	1.45	1.55	1.07	2.34	0.51	0.50
Days Lost or Charged	2592	1220	1107	2816	231	245
Severity Rate	377	172	148	366	30	31

Table 8.2 Injury Record by Divisions - 1967

Division	Medical Treatment Cases	Number of Serious Injuries	Disabling Injuries Number	Freq.	Sev.	Exposure Hours	Exposure Hours Since Last Disabling Injury
Analytical Chemistry	24	3				333,238	5,119,076
Chemical Technology	43	2				453,251	3,914,380
Chemistry	10	1				212,443	540,223
Director's	12					248,196	808,024
Electronuclear	4	1	1	9.28	1625	107,764	86,741
Health Physics	38					409,204	1,432,467
Instr. & Controls	54	4				580,164	1,526,555
Mathematics	5	1				170,570	1,334,819
Metals & Ceramics	41	1				598,540	2,282,669
Neutron Physics	5					145,554	915,279
Physics	8	2				109,979	2,525,132
Reactor	8					113,868	1,085,539
Reactor Chemistry	15	1				198,789	1,407,585
Solid State	10					168,829	1,824,033
General Engineering	9					355,458	1,360,831
Health						62,571	890,894
Isotopes	31	5				297,060	709,758
Laboratory Protection	16	1				158,434	195,529
Operations	82	1				447,808	1,028,319
Personnel	18	1	1	4.60	32	217,177	49,804
Plant & Equipment	1098	64	2	0.88	27	2,269,869	1,221,745
Technical Information	14					250,850	2,018,445
Inspection Engineering	12	1				87,924	604,969
PLANT TOTAL	1557	89	4	0.50	31	8,032,396	1,915,987

Table 8.3 Four-Plant Tabulation of Injuries - 1967

	Labor Hours (Millions)	Disabling				Serious	
		Number of Injuries	Frequency Rate	Days Lost or Charged	Severity Rate	Number of Injuries*	Frequency Rate
ORNL	8.0	4	0.50	245	31	89	11.08
ORGDP	4.7	3	0.63	409	86	36	7.58
Y-12	10.7	6	0.56	549	51	127	11.89
Paducah	2.0	1	0.51	175	89	33	16.79

*Includes the number of disabling injuries.

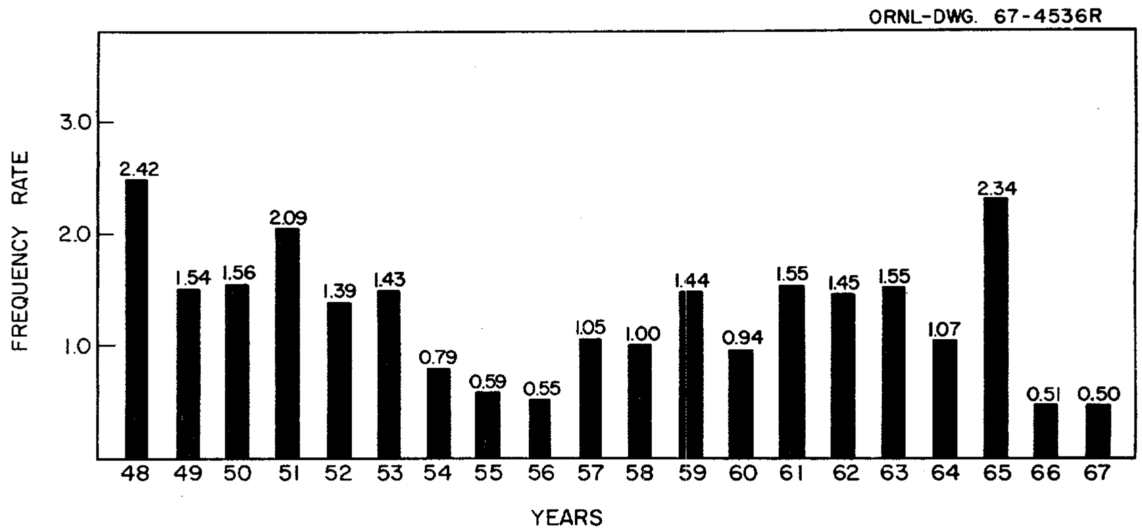

Fig. 8.1 Disabling Injury Frequency Rates Since Inception of Carbide Contract.

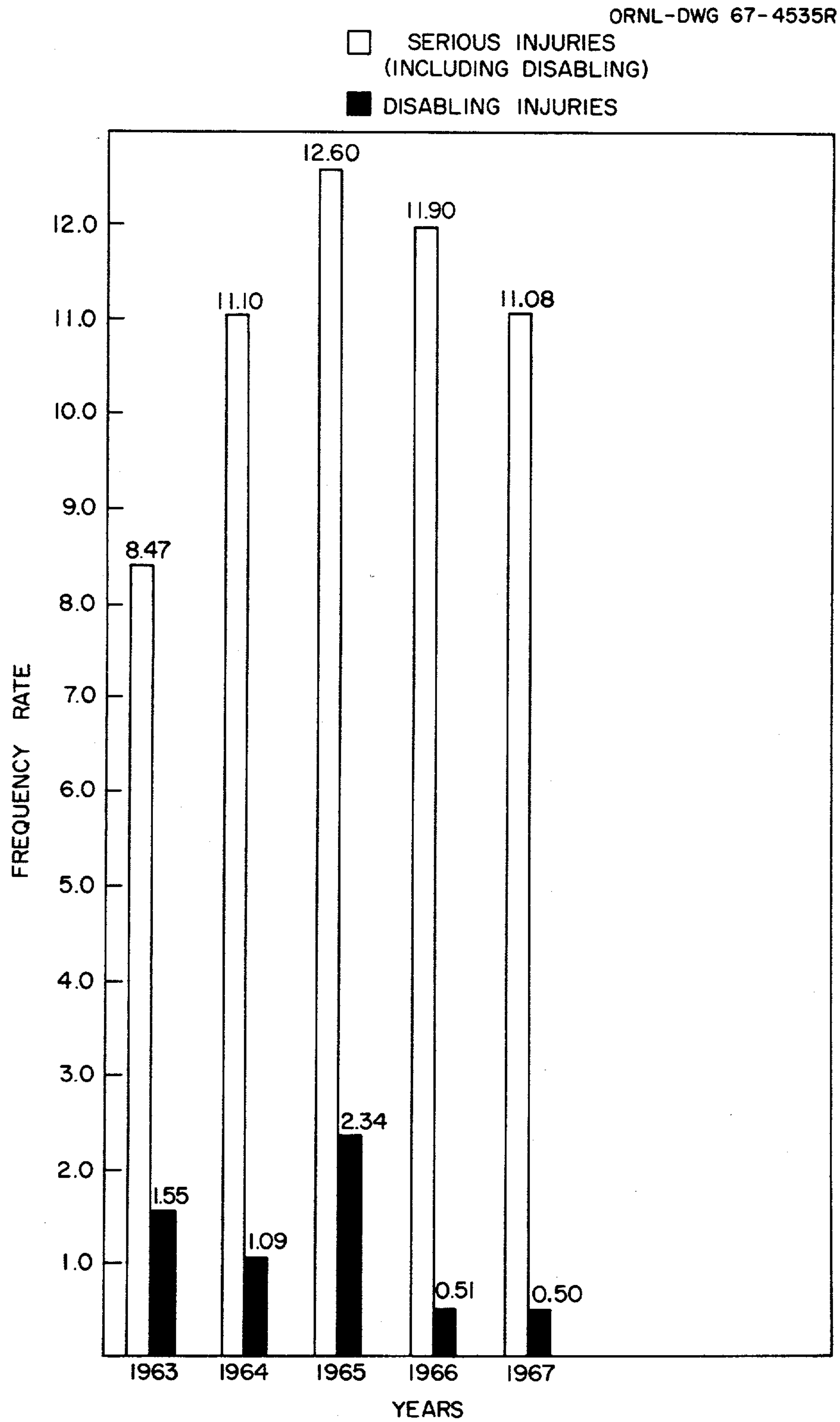

Fig. 8.2 Injury Frequency Rates - 1963-1967.

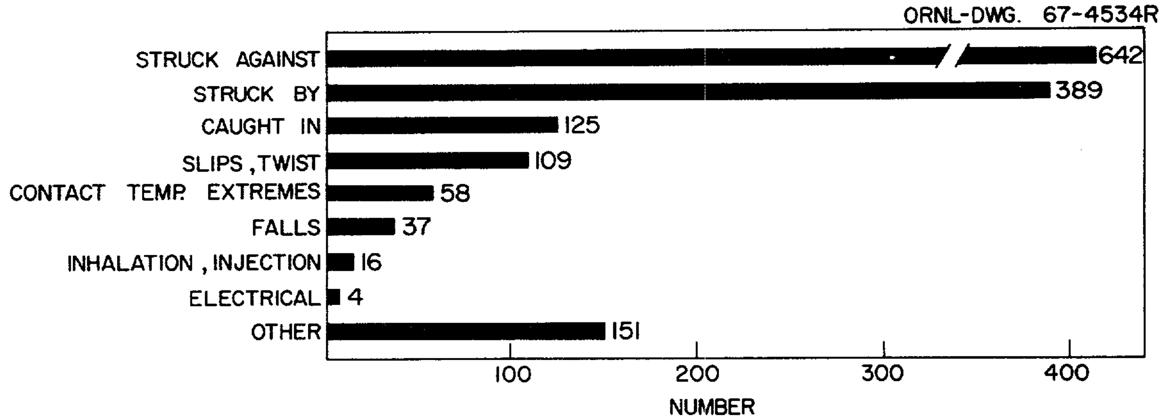

Fig. 8.3 Accidents by Types.

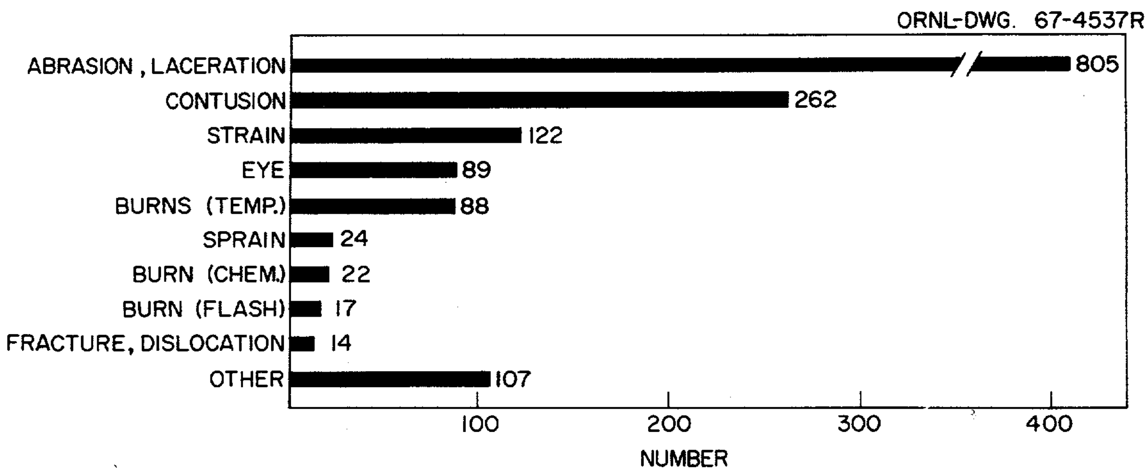

Fig. 8.4 Nature of Injury.

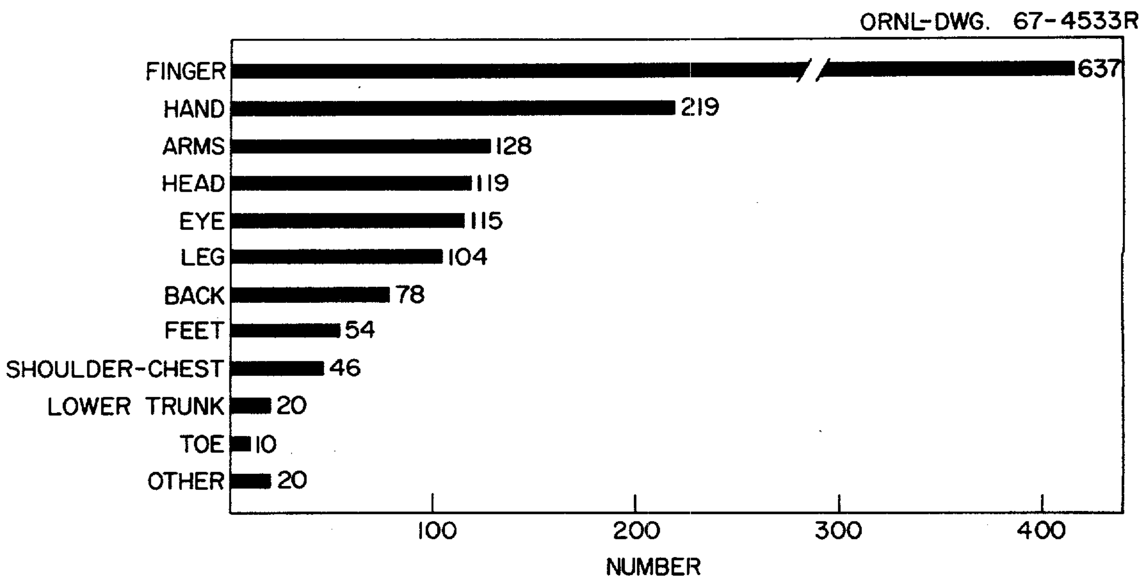

Fig. 8.5 Part of Body Injured.

9.0 LABORATORY ASSAYS

Laboratory Assays Units provide laboratory support to the Health Physics Monitoring Sections. These services include (1) the analysis of body fluids and excreta (bio-assay) for the monitoring of personnel for internal radiation exposure, (2) the radiochemical analysis of environs samples, (3) counting services for the environs monitoring and radiation survey programs, (4) autoradiography, and (5) whole body counting (in vivo gamma spectrometry).

9.1 Bio-Assay Analysis

The number and types of analyses performed by the Bio-Assay Unit during 1967 are given in Table 9.1. A total of 6,069 analyses were performed which include 5,399 analyses on samples submitted by donors and 607 analyses on standard and blank samples analyzed for control purposes. Approximately 87 percent of the samples were analyzed for either the alpha emitters or strontium. The total number of analyses on samples submitted during 1967 was about the same as the number processed during 1966.

9.2 Counting Facility

Over 200,000 samples were processed by the counting facility during 1967. A tabulation of the number and types of samples counted is presented in Table 9.2. The total number of samples processed was about the same as the previous year.

9.3 Environs Monitoring Sample Analysis

Table 9.3 presents the number and type of environs samples analyzed and the type of analysis performed on each type of sample. A total of 5,995 samples was analyzed during 1967 as compared with 8,935 samples analyzed in 1966. The decrease was primarily the result of discontinuing autoradiography of continuous air monitoring filters from the various Health Physics laboratory areas on February 6, 1967. Analysis of environs monitoring samples may range from a single determination to as many as 12 determinations per sample depending upon the radionuclides present. The methods used by the various analytical groups are generally described in the ORNL Master Analytical Manual.

9.4 Autoradiography

There were 1,889 films processed during 1967 in support of radio-particulate studies conducted by the Environs Monitoring Units.[10]

9.5 Whole Body Counter

During the calendar year 1967 the whole body counting program included 983 counts on 843 persons; 813 or approximately 83 percent of these counts

[10] Methods described in ORNL-2601, Radioactive Waste Management at Oak Ridge National Laboratory.

showed a normal human gamma spectrum. Of the 983 counts, 29 were initial counts made on persons involved in possible contamination incidents. All but one of the 29 counts showed evidence of internal contamination present. Seventy-nine counts were made for the purpose of further investigation of positive counts; all but five of these showed evidence of continued internal contamination.

In addition to the whole body counts noted above, two counts made on the wound probe following minor injuries in contaminated areas showed positive evidence of contaminated wounds. Forty-one counts were made for calibration or standard counts, 31 counts were made to identify contamination in air or surface contamination samples, 38 counts were made in an attempt to pinpoint the source of unexplained fluctuations in the natural background in the iron room and 31 counts were made for the purposes of developing and improving in vivo or wound probe counting capabilities.

There was no case, based on data collected by the IVGS, for which the AEC reportable level for occupational workers (one-half of a permissible body burden averaged over the year) was exceeded.

Table 9.1 Bio-Assays Analyses—1967

Analytical Procedure	Number of Analyses	
Urine:		
Trans Pu	427	
Sr	1,948	
U	826	
TRE (total rare earths)	-	
^{3}H	318	
^{137}Cs	175	
^{239}Pu	1,468	
^{106}Ru	44	
^{32}P	-	
Other	169	
Total		5,375
Fecal:		
Gross Alpha	3	
Sr	8	
U	3	
Others	11	
Total		24
Standards and blanks		670
GRAND TOTAL		6,069

Table 9.2 Counting Facility Resume—1967

Types of Samples	Number of Samples			Unit	Weekly
	Alpha	Beta	Gamma	Total	Average
Survey Area Samples					
Smears	77,186	72,126		149,312	2,817.2
Air Filters	25,825	25,825		51,650	974.5
Environs Monitoring					
Air Filters	1,867	1,867		3,734	70.5
Gummed Paper		1,352		1,352	25.5
Rain Water		1,335		1,335	25.2
White Oak Lake Effluent	185	1,098		1,283	24.2
Animal Thyroids			279	279	5.2
Milk			449	449	8.5
GRAND TOTAL	105,063	103,603	728	209,394	3,950.8

Table 9.3 Environmental Monitoring Samples—1967

	Sample Type	Type of Analyses	Number Samples
1.	Monitoring network filters	Gross beta, autoradiogram	1,922
2.	Gummed paper fall-out trays	Gross beta, autoradiogram	1,549
3.	CAM filters*	Autoradiogram	297
4.	Rain water	Gross beta	774
5.	White Oak Dam effluent	Gross beta, radiochemical, gamma spectrometry	436
6.	Clinch River water	Gross beta, radiochemical, gamma spectrometry	20
7.	Raw milk	Radiochemical	455
8.	Pasture grass	Radiochemical, gamma spectrometry	224
9.	Potable water	Radiochemical, gamma spectrometry	12
10.	Silt composites	Radiochemical, gamma spectrometry	27
11.	Animal thyroids	Gamma spectrometry	279
		TOTAL	5,995

*Discontinued autoradiography on 2-6-67.

10.0 HEALTH PHYSICS INSTRUMENTATION

The Health Physics Division shares with the Instrumentation and Controls Division the responsibility for the development of electronic radiation monitoring instruments used in the Laboratory health physics program. Normally the Health Physics Division is responsible for determining the need for new instrument types and modifications to existing types, specifies the health physics requirements for design, and approves the final design. The Health Physics Division is also responsible for calibrating all instruments used in the health physics program and is allocated the funds for maintenance of these instruments. Maintenance is performed or cross-ordered by the Instrumentation and Controls Division.

Non-electronic personnel monitoring devices are designed, tested, calibrated, and maintained by Health Physics Division personnel.

10.1 Instrument Inventory

The electronic instruments used in the health physics program are divided, for convenience in servicing and calibrating, into two classes: the first class includes battery-powered portable instruments; the second class includes the stationary instruments that are AC powered. Portable instruments are assigned and issued to the Radiation Survey Units. Stationary instruments are the property of the Laboratory division which has the monitoring responsibility in the area in which the instrument is located. Table 10.1 lists portable instruments assigned at the end of 1967; Table 10.2 lists stationary instruments in use at the end of 1967. There were net increases in 1967 of 61 portable instruments and 85 stationary instruments.

During 1967, 777 new pocket meters, 481 new fiber dosimeters (200 mR range) and 36 personal radiation monitors (PRM) were issued by ORNL Stores. Most of the pocket meters issued were replacements for instruments which had been lost or damaged.

Inventory and Service Summaries for health physics instruments are prepared on a CDC 1604. These computer programmed reports enable the Instruments Group to maintain a current inventory on most health physics instrument requirements.

The allocation of stationary health physics monitoring instruments by divisions is shown in Table 10.3

10.2 Calibration Facility

The Health Physics Division maintains a calibration facility for the calibration and maintenance of portable radiation instruments and personnel metering devices. The facility is equipped with calibration sources, remote control devices, and shop space for the use of Instrumentation and Controls Division maintenance personnel. Health Physics personnel assign, arrange for maintenance of, calibrate, provide delivery services for, and maintain inventory and servicing data of all portable health physics survey instruments.

Portable instruments should be serviced (1) whenever repairs are needed, (2) at least once each two months for those which have replacement-type batteries, and (3) at least once each three months for those instruments which have "permanent" (rechargeable) batteries. The number of calibrations of portable instruments for 1967 is shown in Table 10.4.

Stationary instruments are calibrated by Calibrations Group personnel or by Radiation and Safety Surveys personnel who use sources which are designed, standardized, and provided by the Calibrations Group.

10.3 Instrumentation Developments and Innovations

Five of the newly designed alpha air proportional instruments, Model Q-2912 (Figure 10.1), have been in use for several months and have functioned rather well.

The plutonium air monitor (Figure 10.2) is described in a report, ORNL-TM-2011 (September, 1967). The prototype model of this air monitor has performed satisfactorily for more than two years.

An audible signal, amplifier-speaker attachment for portable instruments (ORNL Q-2940) was designed. The device is self-powered, quite small, and produces a rather loud chirp for each count when plugged into the earphone jack of an instrument.

The method for charging the nickel-cadmium batteries in portable instruments was modified to eliminate the direct coupling of the instrument to the 110 volt line supply. The modification provides 12 volts, DC, by means of an in line transformer-rectifier. The instrument is thus isolated from the AC supply and the potential for electrical shock is eliminated.

A low level (less than 500 counts per minute) counter, Model Q-3031, for alpha sample analysis was designed for bio-assay use. Six of these counters are mounted in one NIM bin.

Table 10.1 Portable Instrument Inventory - 1967

Instrument Type	Instruments Added 1967	Instruments Retired 1967	Assigned Inventory Jan. 1, 1968
GM Survey Meter	18	1	452
Cutie Pie	18	11	460
Juno	0	0	32
Alpha Survey Meter	25	4	237
Neutron Survey Meter	15	0	88
Miscellaneous	1	0	15
TOTAL	77	16	1284

Table 10.2 Inventory of Facility Radiation Monitoring Instruments for the Year - 1967

Instrument Type	Installed During 1967	Retired During 1967	Total Jan. 1, 1967
Air Monitor, Alpha	16	0	105
Air Monitor, Beta	11	3	188
Hand-Foot Monitor	3	0	33
Lab Monitor, Alpha	25	1	151
Lab Monitor, Beta	34	2	208
Monitron	10	1	245
Other	11	16	132
TOTAL	107	22	1062

Table 10.3 Health Physics Facility Monitoring Instruments Divisional Allocation - 1967

ORNL Division	α Air Monitor	β Air Monitor	α Lab Monitor	β Lab Monitor	Monitron	Other	Total
Analytical Chemistry	4	12	11	17	15	6	66
Chemical Technology	45	50	48	36	38	35	252
Chemistry	9	9	19	25	20	8	90
Metals and Ceramics	12	16	16	17	11	12	84
Reactor	7	12	2	9	11	10	51
Isotopes	19	32	24	41	58	25	199
Operations	1	42	2	19	60	16	140
All Others	7	15	29	44	32	53	180
TOTAL	105	188	151	208	245	165	1062

Table 10.4 Calibrations Resume - 1967

		1966	1967
A.	Portable Instruments Calibrated		
	1. Beta-Gamma	3,792	3,877
	2. Neutron	152	203
	3. Alpha	984	1,063
	4. Pocket Chambers and Dosimeters	3,224	3,815
B.	Films Calibrated		
	1. Beta-Gamma	1,310	1,684
	2. Neutron	20	40

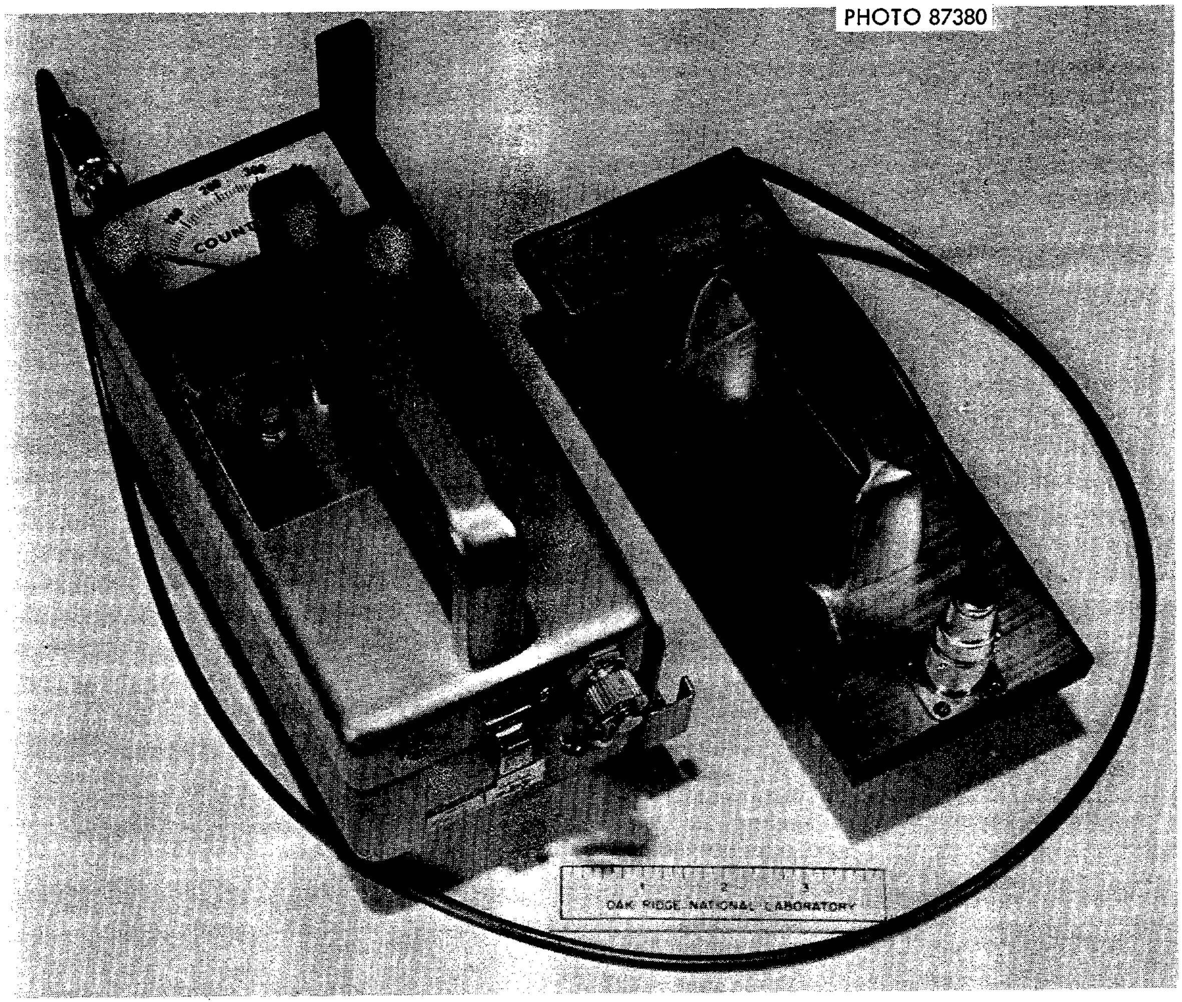

Fig. 10.1 Alpha Survey Meter with Air Proportional Counter, Q-2912

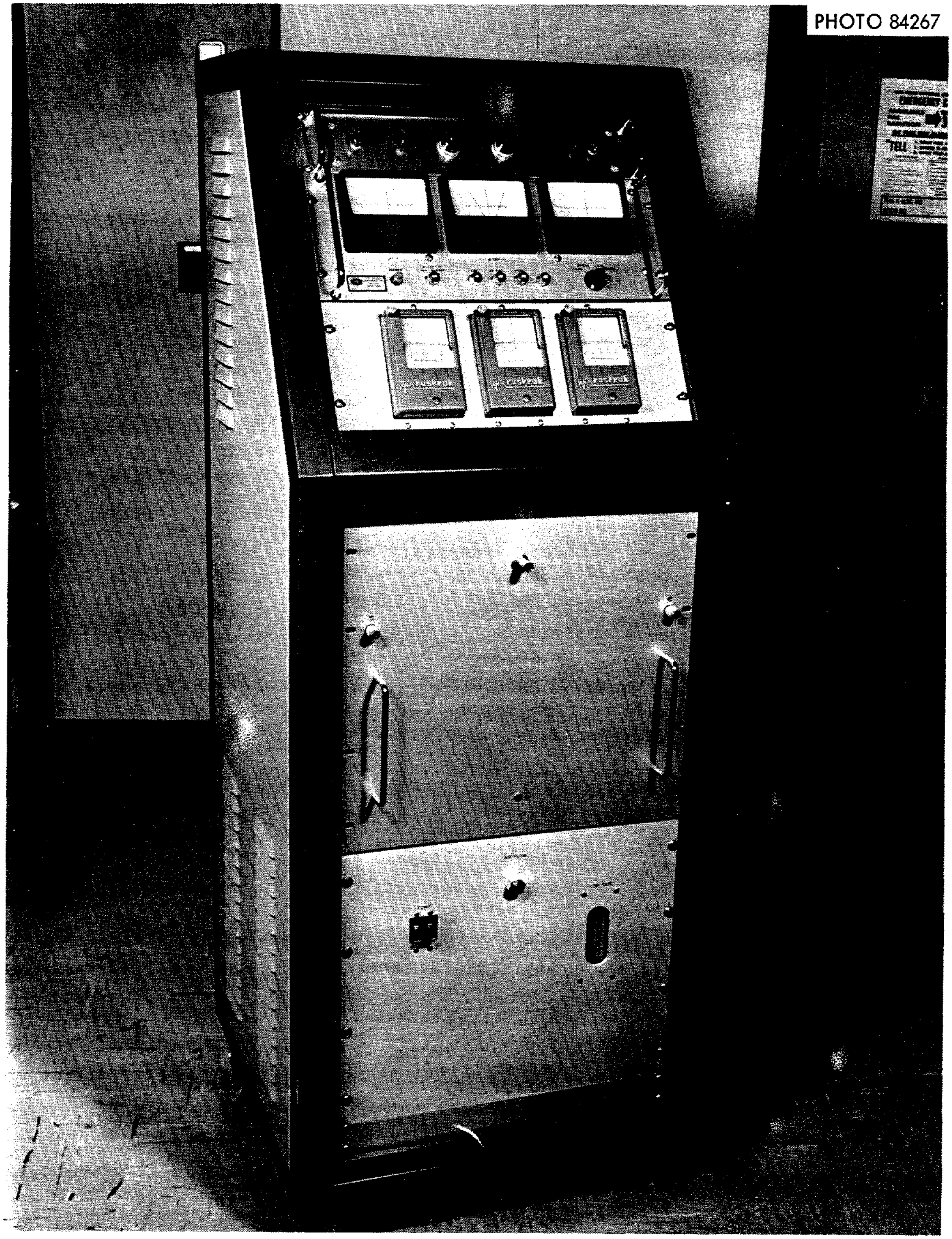

Fig. 10.2 Photograph of Prototype Alpha Air Monitor.

11.0 PUBLICATIONS AND PAPERS

D. M. Davis, "Organization of Radiation Protection Services", paper presented at the Meeting of IAEA Regional Health Physics Study Group, Athens, Greece, October 16-20, 1967.

D. M. Davis, "Education and Training in Radiation Protection", paper presented at the Meeting of IAEA Regional Health Physics Study Group, Athens, Greece, October 16-20, 1967.

D. M. Davis, "Radiation Monitoring at Research Reactors", paper presented at the Meeting of IAEA Regional Health Physics Study Group, Athens, Greece, October 16-20, 1967.

D. M. Davis, Report of Trip to Athens, Greece, by D. M. Davis during the Period October 16-20, 1967, ORNL-CF 67-11-72, November 10, 1967.

D. M. Davis, Health Physics and Safety Annual Report for 1966, ORNL-4146, August, 1967.

L. B. Farabee, "Improved Procedure for Radio Strontium Analysis of Human Urine", Proceedings of the Twelfth Annual Bio-Assay and Analytical Chemistry Meeting, Gatlinburg, Tennessee, October, 1966, CONF-661018, 1967.

E. D. Gupton and D. M. Davis, "Health Physics Instruments", Chapter 15 of Principles of Radiation Protection, Edited by K. Z. Morgan and J. E. Turner, John Wiley & Sons, Inc., New York (1967).

E. D. Gupton, Alpha Air Monitor for ^{239}Pu, ORNL-TM-2011, September 27, 1967.

J. C. Hart, "On the Legalistic Aspects of the Radiation Exposure Record", Health Physics, Vol. 13, pp. 319-26, 1967.

J. C. Hart, "Techno-Legal Factors to be Considered in Developing the Radiation Dose Record", paper presented at the Thirteenth Annual Bio-Assay and Analytical Chemistry Meeting, Lawrence Radiation Laboratory, Berkeley, California, October 12-13, 1967.

J. C. Hart, "Medico-Legal Aspects of TLD", paper presented under auspices of Blue Grass Chapter of HPS, et al., at the Symposium on Thermoluminescence Dosimetry, October 28, 1967.

H. C. Hoy, "Notes of Panel Discussion", p. 387, Proceedings of the Engineering Problems of Controlled Thermonuclear Research Symposium, Gatlinburg, Tennessee, October, 1966, CONF-661016, 1967.

G. D. Kerr and D. R. Johnson, "Comparison of Radiation Dosimeters at the Health Physics Research Reactor", paper presented at the Twelfth Annual Meeting of the Health Physics Society, Washington, D. C., June 18-22, 1967.

G. D. Kerr and J. S. Cheka, "Developments in Metaphosphate Glass Dosimetry", paper presented at the First Health Physics Society Midyear Topical Symposium on Personnel Radiation Dosimetry, Chicago, Illinois, January 30 - February 1, 1967.

12.0 VISITORS AND TRAINING GROUPS

During 1967 there were 106 visitors to Health Physics and Safety, as individuals or in groups, for training purposes. This figure represents an increase of 30 percent over the number of visitors recorded during 1966. Table 12.1 is a listing of the training groups which consisted of six or more persons.

Table 12.1 Training Groups in Health Physics and Safety Facilities during 1967

Facility	Number	Training Period
U. of Arkansas (Radiological Health)	6	4/26/67 - 4/28/67
AEC Fellowship	13	6/14/67 - 8/25/67
U. of North Carolina (Public Health)	8	9/5/67 - 9/8/67
ORAU 10-Weeks Course	13	10/23/67 - 10/27/67

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